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ESARDA
European Safeguards Research and Development Association

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SESSION 1

OPENING

Opening Address by the President of ESARDA
The 31st ESARDA Annual Meeting: ESARDA 40 years
26 May 2009, Vilnius

Vice Minister of the Ministry of Energy,
Dear Participants,
Dear Colleagues,
Ladies and Gentlemen,

My name is Elina Martikka. It is my honour and great pleasure, as the President of ESARDA, to open the 31st ESARDA Annual Meeting. This Annual Meeting is a special one because ESARDA celebrates its 40th anniversary this year. It is a great pleasure to have our special meeting here in Vilnius, the European Capital of Culture in 2009. This is the first ESARDA meeting held in a relatively new EU Member State. I am happy to note that the new Member States seem to be active in hosting the meetings in the future too. Namely, the ESARDA 2011 annual meeting will be hosted by our Hungarian colleagues in Budapest. But now, many thanks to Lithuania, and VATESI in particular, for their initiative in hosting this meeting.

I would like to add that this ESARDA meeting is well timed. Yesterday, during the night between Sunday and Monday, North-Korea carried out its second nuclear test. The nuclear test and matters relating to it belong in ESARDA's field of activities; this nuclear test will certainly be one of the most discussed topics at this meeting. India and Pakistan carried out nuclear tests during ESARDA's meeting in Helsinki 1998. It is our hope that, in the future, a habit will not be made out of carrying out nuclear testing at same time with ESARDA's meetings. Should this become the case, we must consider a reduction in the number of meetings held.

During its 40-year history ESARDA has been constructing the safeguards world and has reached a remarkable position in the international safeguards community. Success has followed from responding to the needs of our customers. How can we measure our success? One indicator is that the number of members in our association has increased during these years. ESARDA members number is 28 today. There were eleven members 10 years ago. We also have members from non-EU-Member States like Norway and Switzerland. The newest EU Member States, who joined the EU in 2004, like Lithuania, Hungary, the Czech Republic and Romania, have found ESARDA and its activities soon. It is expected that the number of ESARDA individual members will increase in the years to come. Today ESARDA seems to be a desired association to join in.

It is worth recognizing today how much and how quickly the world is changing. These are crucial times - in terms of both concern and hope. We live in a world where, if you have nuclear weapons, you have power and insurance against attack. A week ago the Director General of the IAEA, Mr. ElBaradei, said in an interview that he is worried about the new type of nuclear weapons states, the so called virtual nuclear weapons states. These states have the knowledge, technology and materials to prepare a nuclear weapon in a few months, if necessary. Mr. ElBaradei said that the number of virtual weapons states could be 10 to 20.

The situation in Pakistan is terrifying. The Pakistan army is at war against the Talibans in the middle of 60 nuclear warheads. If things go badly what will happen to the nuclear warheads? Will terrorists get hold of them?

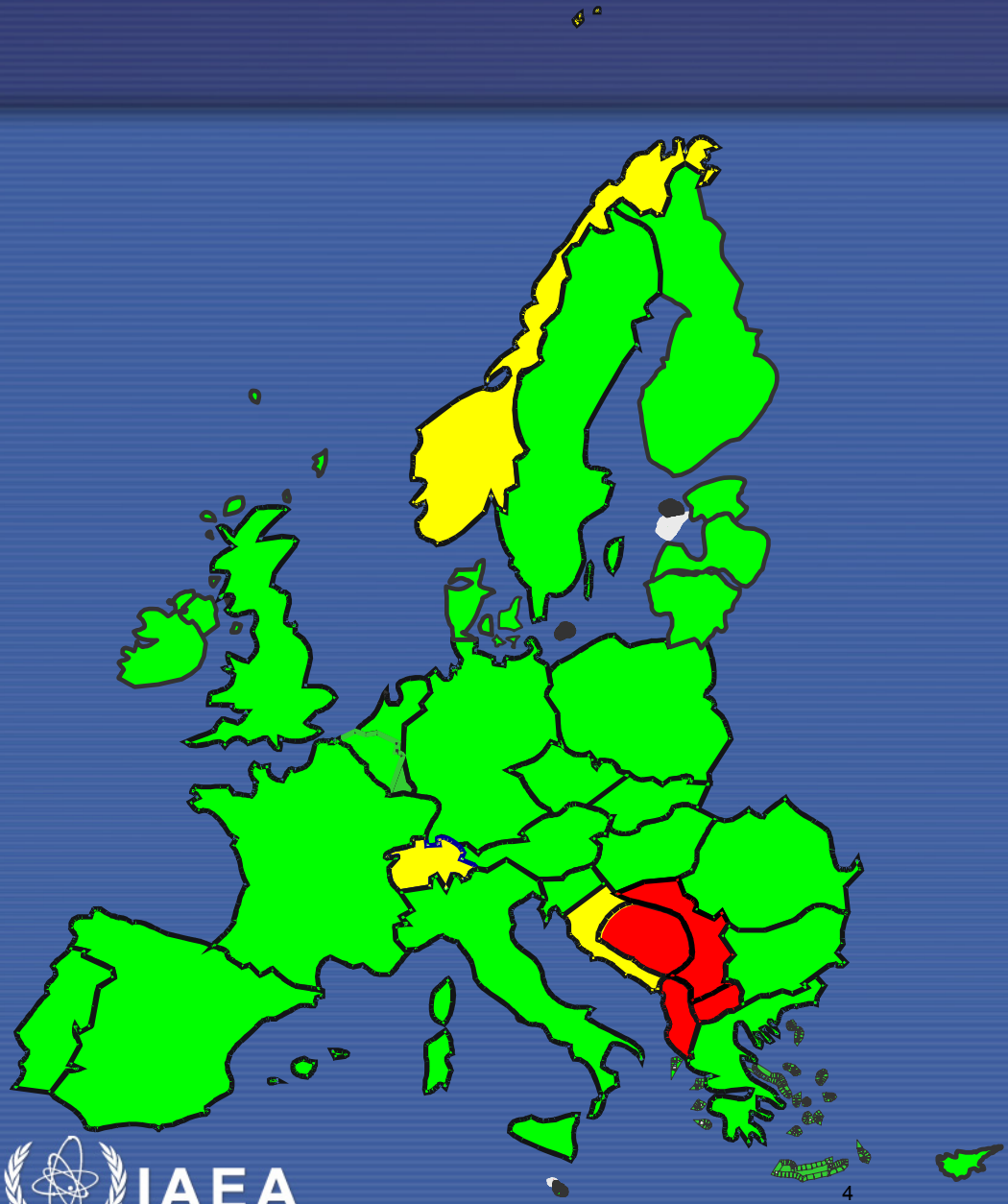
What will happen to the NPT? The NPT review conference will be held in April 2010. Before that it would be very important to find a way for the succesful continuation of the NPT and the effective disarmament of nuclear weapons of which there are 24 000 in all, about 200 of which are owned by “de facto” nuclear weapons states like India, Pakistan, Israel and North Korea.

Great hopes for improving the situation are being invested on the ongoing negotiations between the superpowers, the USA and the Russian Federation. Hopefully a good example that points towards a peaceful future without nuclear weapons will be found. The first concrete step towards peace would be to bring into force the CTBT and to maintain the NPT as powerful.

To be successful in the future, we must be sensitive in recognizing changes and challenges in our operational environment and in understanding how our customers and their needs are developing and changing.

During these busy symposium days we will have interesting presentations about Integrated Safeguards, national safeguards systems, developments in verification issues, issues related to illicit nuclear trafficking etc. So, it is necessary to have parallel sessions in order to have as many valuable and interesting presentations as possible. And please take the opportunity to make use of the poster session where you can discuss with the authors. I hope that we will remember during these busy and interesting days that this is not only a meeting with presentations but also a forum for meeting old friends and finding some new. Therefore, please be active, communicate, learn and try also to find some time to enjoy the beautiful Vilnius.

Status of Additional Protocols in Europe



- EU States with AP in force
- Non-EU States with AP in force
- Non-EU States without AP

- Status of Integrated Safeguards Implementation in the EU as of May 2009

Austria	2008	Italy	2008
Bulgaria	2005	Latvia	2006
Czech Rep.	2007	Lithuania	2007
Denmark	2009	Luxembourg	2009
Estonia	2009	Malta	2008
Finland	2008	Poland	2006
Greece	2008	Portugal	2008
Hungary	2004	Romania	2007
Ireland	2008	Slovenia	2005
		Sweden	2009

Status of Integrated Safeguards Implementation in the European Union as of May 2009

Integrated safeguards under consideration

Belgium	**2nd Quarter 2009**
Netherlands	**2nd Quarter 2009**
Slovakia	**2nd Quarter 2009**
Spain	**3rd Quarter 2009**
Germany	**4th Quarter 2009**
Cyprus	**Pending Broader Conclusion**

Recent LLLC Agreements

IAEA/Euratom Lower Level Liaison Committee
Recent products of the LLLC include:

•Agreed Partnership Approach paper on integrated safeguards for gas centrifuge enrichment plants

•Agreed Partnership Approach papers on short-notice random inspection (SNRI) systems for two fuel fabrication plants:
oVasteras plant, Sweden
oDessel plant, Belgium

Areas of Current LLLC Discussion

- Current LLLC projects include:
- Reaching final agreement on SNRI Partnership Approach papers for additional fuel fabrication plants:
  - Lingen plant, Germany
  - Juzbado plant, Spain
- Establishing Partnership Approach papers for on-load reactor sites
- Continuing to improve joint-use equipment practices and joint training

SESSION 2

FUTURE CHALLENGES

A NONPROLIFERATION IMPACT ASSESSMENT OF THE GNEP ALTERNATIVES

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I. INTRODUCTION

Since the beginning of the atomic era, the dual nature of nuclear energy has been clear. The Global Nuclear Energy Partnership (GNEP) proposal aimed to support the expansion of nuclear energy while reducing the associated nuclear waste impacts and the risks of nuclear proliferation. Under the Obama Adminstration, the U.S. Department of Energy (DOE) terminated the domestic elements of the GNEP program. In place of plans for near-term commercial deployment, DOE will focus on research and development for the U.S. domestic nuclear fuel cycle, with a focus on improving "proliferation resistance," as part of the Advanced Fuel Cycle Initiative (AFCI). The United States continues to take part in the international component of the GNEP program, while reviewing how best to achieve its international objectives. The Obama Administration has also proposed to eliminate the Yucca Mountain repository program, while exploring alternatives for nuclear waste disposal and continuing participation in the repository license proceeding before the U.S. Nuclear Regulatory Commission..

It is important for policy makers to understand the nonproliferation impacts of the programs that are designed to influence the international nuclear fuel cycle. The draft Nonproliferation Impact Assessment (NPIA)[1] is a companion document to the draft Programmatic Environmental Impact Statement (PEIS)[2] for the GNEP program that was released 17 October 2008 for public comment. It compares the proliferation risks and nonproliferation benefits of the alternatives considered under the GNEP program in the PEIS. It also focuses on how each of these alternatives may advance nonproliferation policy objectives. While a decision on the GNEP PEIS is under review, the NPIA has value on its own, both in helping to frame consideration of spent fuel management options and in setting the stage for further work on proliferation risk assessment and for potential future nonproliferation impact assessments.

This assessment addresses proliferation risks from the nuclear fuel cycle. "Proliferation resistance" is a system concept that considers combinations of technologies and other measures that work together to impede proliferation. No single feature can prevent diversion or misuse of nuclear energy. Therefore, robust safeguards and physical security will be required for civilian use of nuclear energy,. The GNEP program provides the opportunity to advance the following nonproliferation and international security policy objectives:

- Limit the further spread of enrichment and reprocessing programs, by providing reliable fuel services as a viable alternative to such programs;
- Halt the build-up and eventually draw down stocks of separated plutonium;
- Develop and promote reactors and fuel cycles with reduced proliferation and security risks; and
- Improve international safeguards approaches to verify that countries are not misusing nuclear energy for weapons purposes.

The focus of this nonproliferation impact assessment is on policy issues, with the Proliferation Resistance and Physical Protection (PR&PP) methodology used primarily in a supporting role to clarify technical nonproliferation issues. This assessment is intended for information purposes, to frame a full and open discussion of the nonproliferation issues. It does not represent any decision or select a preferred alternative.

The alternatives for the U.S. domestic nuclear fuel cycle identified in the PEIS vary widely in reactor type, fuel type, and processing of spent fuel. For the purposes of this assessment, these alternatives fall into three broad categories:

- Once Through Fuel Cycles: In these alternatives, fuel is used once in reactors, after which spent fuel is treated as waste, and is stored and eventually disposed of in a geologic repository. The alternatives under consideration use uranium-based or uranium/thorium-based fuel in thermal-neutron reactors (light-water reactors), heavy water reactors or high-temperature gas-cooled reactors.

- Full Actinide Recycle: In these alternatives, spent fuel from light-water reactors is reprocessed, and transuranic elements are removed from the waste stream and recycled. These alternatives minimize long-term waste hazards and treat spent fuel as a resource. The alternatives under consideration use fast reactors and LWRs.

- Partial Actinide Recycle: In these alternatives, spent fuel from light-water reactors is recycled in thermal-neutron reactors. Some of the transuranic elements are removed from the waste stream and recycled, but a significant portion remains in the waste stream. The alternatives in this category also include deep burn HTGR recycle of reprocessed transuranic elements, and DUPIC.

II. PROLIFERATION RISK and its ASSESSMENT

The risk of nuclear proliferation is inherent in the use of nuclear energy. The same underlying fission process that is used in nuclear power plants to produce electricity can also be used in nuclear weapons to produce explosions. The principal risks of nuclear proliferation come from the characteristics of the nuclear fuel cycle and how the fuel cycle is organized internationally. Fuel for the U.S. fleet of 104 LWRs must be enriched to a uranium-235 (U-235) concentration of 3-5%, which is far too low for use in a nuclear weapon. However, the *enrichment* processes used in LWR fuel production can also be used to produce high-enriched uranium for use in weapons.

Use of nuclear fuel in a reactor generates plutonium, which also can be used both in reactor fuel and in nuclear weapons. Both uses require *reprocessing* of the spent reactor fuel. Thus, recycling plutonium at the back end of the fuel cycle carries similar proliferation risks to the enrichment of uranium at the front end.

Proliferation challenges arise in the context of both states and non-state actors, though the different aims and capabilities of these two types of actor lead to different types of risk. These challenges make it clear that improved mechanisms for impeding the spread of enrichment and reprocessing capabilities and for detecting and deterring proliferation or nuclear terrorism are needed.

One approach for reducing proliferation risk is to seek to build "proliferation resistance" into both nuclear facilities and the structure of the international nuclear fuel cycle. It is important to recognize that it is not possible to make any system completely "proliferation proof." Rather, the term "proliferation resistance" should be understood, and is used in this assessment, as a systems approach to reduction of risk through a combination of intrinsic features (*e.g.*, physical and engineering features of a nuclear energy technology) and extrinsic measures (*e.g.*, safeguards and physical barriers) that make it harder to divert or misuse nuclear materials or facilities without detection. For example, since proliferation resistance is a system of features, not a single feature, it is obvious that no simple technical change, like blending minor actinides into plutonium, can have a substantial impact on the proliferation resistance of reprocessing.[3] Proliferation resistance also includes institutional arrangements that may deter states from proliferation or reduce incentives to pursue sensitive technologies.

This draft NPIA draws on nonproliferation objectives of the U.S. Government as the basis for a *policy evaluation* of proliferation risk, and on the Proliferation Resistance and Physical Protection (PR/PP) methodology as the basis for a *technical evaluation* of proliferation risk. However, because of its broad, programmatic nature, the GNEP PEIS does not include all the technical detail necessary to evaluate fully the potential nonproliferation impacts of the identified alternatives. Those details include the specific technologies to be deployed, the timing and scale of that deployment, and other important technical details, such as the form and quantity of fuel recycled, the location of facilities, and transport of nuclear materials, including spent and recycled fuel. Future assessments of nonproliferation impact may be necessary to inform decisions on implementing any of the alternatives and necessary as both research and development programs and the PR/PP methodology evolve.

III. NONPROLIFERATION OBJECTIVES

The advancement of nonproliferation and nuclear security interests requires a combination of policy and technological innovation. Analysis limited to policy alone may make unrealistic assumptions of what is technically feasible, whereas analysis limited to technology alone may overlook opportunities for institutional arrangements that reduce proliferation risks. This draft NPIA attempts to blend those elements through what is termed a "policy effects" analysis. It evaluates objectives

that may have significant nonproliferation impact and compares technical proliferation risks and nonproliferation benefits in the GNEP PEIS alternatives.

Specific nonproliferation policy objectives include:

1. *Limiting the further spread of enrichment and reprocessing*. In support of this objective, the United States is considering concepts for comprehensive fuel services arrangements as an alternative for countries that might otherwise consider developing their own uranium enrichment or spent fuel reprocessing facilities. This concept would include assured supply of fresh fuel, but the key innovation is that it also contemplates return of the spent fuel from the recipient country, though not necessarily to the same country that supplied the fresh fuel.
2. *Halting the build-up and eventually drawing down stocks of separated plutonium*. Current stocks of separated civil plutonium total roughly 250 metric tons worldwide, largely as a result of an imbalance between the reprocessing and reuse of such materials in reactors. The extent to which a fuel cycle alternative avoids accumulating separated plutonium or other weapons-usable material and whether it facilitates the drawdown of existing stocks of separated plutonium are considered.
3. *Developing and promoting reactors and fuel cycles with reduced proliferation and security risks*. Mitigating this risk depends on both the technologies and materials used and arrangements for their use or transport. The anticipated expansion of nuclear energy offers the opportunity to develop and deploy innovative nuclear reactors, fuels, and related technologies with increased proliferation resistance.
4. *Improving international safeguards approaches to verify that countries are not misusing nuclear energy for weapons purposes.* GNEP safeguards activities focus on the new types of reactors and spent fuel recycling facilities contemplated under the GNEP program, as well as improving the effectiveness and efficiency of the application of safeguards, export controls, and physical protection.

Because the nonproliferation objectives of GNEP are inherently international in scope and intended impact, the draft NPIA considers additional U.S. policy priorities, including the U.S. ability to influence the nuclear fuel cycle decisions of other countries and the ability to strengthen international safeguards.

International efforts to prevent proliferation have resulted in the current system of interlocking international obligations and national commitments that make up the global nonproliferation system. This includes:

- International treaties, particularly the Nuclear Non-Proliferation Treaty (NPT);
- International organizations, particularly the International Atomic Energy Agency (IAEA);
- Domestic laws, in the U.S. case particularly the Atomic Energy Act of 1954 and the Nuclear Non-Proliferation Act of 1978;
- Export control measures, enforced through national laws, regulations, and the policies articulated in Nuclear Suppliers Group Guidelines;
- International safeguards, implemented by the IAEA in cooperation with its member states, to verify that nuclear material is not diverted from peaceful uses; and
- Physical protection, particularly the Convention on the Physical Protection of Nuclear Materials and Facilities and international guidelines adopted by the IAEA, as well as national measures to secure nuclear material and facilities from threats posed by non-state actors.

A key aim of nonproliferation policy looking forward will be to limit the further spread of enrichment and reprocessing technologies. Comprehensive fuel services arrangements would offer countries a viable alternative to the development of indigenous nuclear fuel cycle programs. Arrangements that include the removal of spent fuel from a recipient country could be a significant incentive for that country to refrain from developing both enrichment and reprocessing capabilities. However, this benefit would need to be balanced against possible negative outcomes.

IV. COMPARISON OF ALTERNATIVES

Table 1 presents a side-by-side descriptive comparison of the three categories of fuel cycle alternatives, based on the policy factors outlined above. The assessments focus on how U.S. domestic fuel cycle choices affect the U.S. ability to influence the fuel cycle choices of other countries, and are based on a combination of technical analysis and expert judgment.

IV.A. Direct Impact on Fuel Cycle

This factor addresses the ability of the United States to affect other countries' fuel cycle practices, particularly by participating in comprehensive fuel services arrangements to discourage the spread of enrichment and reprocessing. Having the capability to participate in such arrangements by accepting other countries' spent fuel for reprocessing and disposal, i.e., back-end fuel services, would offer two significant nonproliferation benefits: (1) it would provide the basis for comprehensive fuel services as an incentive for countries to refrain from sensitive fuel cycle activities, and (2) it would reduce the latent proliferation risk of spent fuel by drawing down and consolidating accumulations worldwide.

By this measure, the full actinide recycle alternatives have the advantage that they would provide the strongest basis for U.S. participation in back-end fuel services. Specifically, these alternatives would reduce technical barriers – and so could also reduce political barriers – to offering back-end fuel services by reducing long-term waste hazards dramatically. Removing plutonium and the other transuranic elements of spent fuel (except for minor process losses) from the waste stream would cause the radiotoxicity of the resulting waste to fall to the level of natural uranium in ore within about 500 years, compared to over 250,000 years for the no action alternative.

Partial actinide recycle alternatives would not achieve as dramatic long-term radiotoxicity reductions but could provide a basis for treating foreign spent fuel as an energy resource. This could help reduce barriers to U.S. participation in back-end fuel services, particularly if accompanied by acceptable arrangements to store and/or dispose of high-level waste from reprocessing outside the United States.

Conceptually, a once-through fuel cycle could also permit acceptance of foreign nations' spent fuel, but only for geologic disposal or storage pending availability of a disposal facility. That approach would be feasible technically but would be limited by policy and legal constraints and may face challenges in gaining public acceptance.

Compared to either the full or partial actinide recycle alternatives, a once-through fuel cycle would also create the greatest demand for uranium enrichment services, which could create incentives for countries to develop indigenous enrichment capabilities. One way to reduce this risk would be to promote a major expansion of enrichment capability in countries that currently provide international enrichment services.

IV.B. Policy Impact on Fuel Cycle

This factor addresses the ability of the United States to influence other countries' fuel cycle policies. The once-through fuel cycle aims to achieve this by demonstrating that reprocessing is not a necessary part of the civilian fuel cycle. A DUPIC fuel cycle, which recycles with only limited separation of fission products, may offer the same benefit. Proponents of a once-through fuel cycle point out that since the United States adopted this fuel cycle in the 1970s, no additional countries have started reprocessing programs for civilian purposes, and several countries have ended their reprocessing programs. However, U.S. opposition has not slowed large-scale reprocessing programs in Europe, Japan, and Russia and has limited U.S. opportunities to influence how those programs are carried out.

Economics has also played a role in limiting the spread of reprocessing capability. The historically low price of uranium and high capital cost of large-scale reprocessing have made it difficult for recycle to be cost competitive. If the price of uranium remains high, or if other factors such as energy security or waste management are seen as justifying the cost, then economics may no longer be sufficient to discourage the spread of reprocessing. In this circumstance, the continuing concerns in the United States regarding spent fuel management and disposition may raise questions internationally about whether a once-through fuel cycle is sustainable.

Existing spent fuel reprocessing programs are based on partial actinide recycling processes that offer relatively limited energy security or long-term radiotoxicity reduction benefits to offset other attributes such as proliferation risk. By pursuing full actinide recycle, the United States could set a higher standard for international reprocessing services. Undertaking a

TABLE 1 Policy Assessment of GNEP Programmatic Alternative

Nonproliferation Impacts	Once-Through Fuel Cycle	Full Actinide Recycle	Partial Actinide Recycle
Fuel Cycle *Direct influence on international comprehensive fuel services and incentives to refrain from enrichment and reprocessing*	*Lowest* influence: *Greatest* barrier to U.S. participation in comprehensive fuel services. Spent fuel has the highest long-term radiotoxicity. *Higher* demand for enrichment services.	*Highest* influence: *Lowest* barrier to U.S. participation in comprehensive fuel services. Minimizes long-term radiotoxicity and treats spent fuel as energy resource. *Lower* demand for enrichment services.	*Intermediate* influence: *Intermediate* barrier to U.S. participation in comprehensive fuel services. Treats spent fuel as energy resource, but offers limited reduction in long-term radiotoxicity. *Lower* demand for enrichment services.
Fuel Cycle *International fuel cycle policy*	*Mixed*: Could reinforce efforts to discourage reprocessing, but unlikely to influence countries that are already pursuing reprocessing.	*Mixed*: Could encourage a major transformation in how commercial reprocessing is carried out, but could reduce ability to argue that reprocessing is unnecessary.	*Lower*: Could encourage an incremental change in commercial reprocessing practices, but could reduce ability to argue that reprocessing is unnecessary. *Mixed* for DUPIC.
Fuel Cycle *Inherent proliferation risk of technology*	*Lowest* risk: Does not create technical basis for separating weapons-usable material from spent fuel. HWRs have *higher* risk of misuse.	*Highest* risk: Capable of separating weapons-usable material, though some modification may be needed depending on the separations technology used. [Note: we omitted the potential for misuse/redesign of fast reactors - too late for this paper.]	*Highest* risk: Capable of separating weapons-usable material, though some modification may be needed depending on the separations technology used. DUPIC has *low* risk because it involves limited separation, although HWRs have *higher* risk of misuse.
Plutonium Stocks	*No improvement*: Reactors could use existing plutonium stocks in MOX fuel, but are not currently licensed to do so.	*Potential significant reduction*: Recycling creates a market for plutonium-bearing fuel. Fast reactor startup would require large quantities of plutonium.	*Potential reduction*: Recycling separated plutonium creates a market for plutonium-bearing fuel. DUPIC alternative is *no improvement* since it would not reduce separated plutonium stocks.
Material Attractiveness	*Lowest*: Spent fuel is bulky and highly radioactive, though radioactivity decays over many decades. HWR spent fuel bundles are smaller and radioactivity decreases more quickly. HTGR spent fuel is difficult to reprocess. Enrichment of fresh fuel varies, but all is LEU.	*Highest*: Removal of fission products and separation of actinides greatly reduces barriers to theft, misuse, or further processing, even without separation of pure plutonium. Fast reactor fuels have higher concentration of weapons-usable materials.	*Highest*: Removal of fission products and separation of actinides greatly reduces barriers to theft, misuse, or further processing, even without separation of pure plutonium. Thermal reactor fuels have lower concentration of weapons-usable materials. *Low* for DUPIC, which has limited removal of fission products.
Safeguards	*Lowest* cost and difficulty: Spent fuel assemblies can be tracked as items. Costs are significantly *higher* for HWRs. *Low to medium* cost and difficulty for thorium or HTGR, which require new safeguards approach.	*Highest* cost and difficulty: Separation processes require continuous monitoring against diversion and novel bulk materials present new measurement challenges. Novel processes may provide new opportunities to detect misuse.	*High* cost and difficulty: Separation processes require continuous monitoring against diversion; bulk material measurement presents familiar challenges. A safeguards approach has been developed for DUPIC.

partial recycle alternative in the United States would not set this higher standard, although it could avoid separation of pure plutonium.

IV.C. Inherent Proliferation Risk of Fuel Cycle Technology

This factor addresses the risk that technologies used in the civil nuclear fuel cycle could be used, either directly or with modifications, to support production of weapons-usable materials. The greatest proliferation risks associated with proposed technologies are those related to chemical separation of spent fuel through reprocessing. All the recycle alternatives aside from DUPIC have this inherent risk, although the cost, difficulty and detectability of any needed process modifications vary. Some of the reactor types involved also pose risks of misuse, including the misuse of heavy water reactors or potential modification of fast reactors into breeders to produce weapons-grade plutonium in their blanket fuel materials.

IV.D. Plutonium Stocks

This factor addresses the ability to avoid accumulations of separated plutonium and to draw down existing stocks worldwide. Any reprocessing of spent fuel can result in an accumulation of separated plutonium or other relatively attractive weapons-usable materials. This is particularly true if economic incentives continue to favor use of fresh uranium fuel.

The full actinide recycle alternatives all involve the use of fast reactors, which require large quantities of plutonium or other fissile material for their initial core loads. The deployment of even modest numbers of fast reactors offers the possibility of converting large quantities of separated plutonium into irradiated fuel and the greatest potential for reducing stocks of plutonium. Thermal neutron reactors can also use plutonium in mixed oxide (MOX) fuel (although most U.S. reactors are not currently licensed to use MOX).

A partial actinide recycle alternative could overcome this hurdle and facilitate the drawdown of existing plutonium stocks by creating a market for MOX fuel. The once-through and DUPIC fuel cycles would neither help nor hinder the drawdown of existing plutonium stocks.

IV.E. Material Attractiveness

This factor addresses how attractive the materials used in the fuel cycle would be for use in nuclear weapons.[3] In this case the assessment is based on the most attractive materials in the fuel cycle alternative, since those are the most likely targets for theft or diversion.

For once-through fuel cycles, the weapons-usable fissionable materials in spent fuel are intermingled with uranium and highly radioactive fission products in large, bulky fuel assemblies. However, over a period of many decades the radioactivity decays and the attractiveness increases, so that an accumulation of spent fuel in a growing number of countries represents a latent proliferation risk that grows over long periods of time. If spent fuel is reprocessed the resulting plutonium-bearing product (whether pure plutonium or a blend of plutonium with other transuranics) is much more attractive. When recycled as reactor fuel, this material is typically diluted with uranium (or possibly an inert material such as zirconium), which can reduce its attractiveness significantly, depending on the relative concentrations and ease of separation of these materials.

Fast reactor fuels require a higher concentration of fissile isotopes than thermal reactor fuels and so are typically more attractive. The DUPIC fuel cycle does retain a relatively high fraction of the fission products with the actinides, leaving the material highly unattractive, but it does convert it from large fuel assemblies to a bulk powder that, while still highly radioactive, is somewhat easier to handle.

IV.F. Safeguards

This factor considers how difficult it is for international safeguards to verify that nuclear materials are not diverted and that facilities are not misused.

Safeguards approaches for the once-through fuel cycle are well known and generally considered effective; enrichment plants pose a significant challenge, but are a common feature of all the alternatives. By contrast, safeguarding spent fuel reprocessing facilities is expensive and very challenging technically for large throughput facilities. Compared to the standard plutonium and uranium solvent extraction process (PUREX) in widespread use that has demonstrated safeguards approaches, many of the advanced separation processes under consideration pose new technical challenges for safeguards, but also hopefully new opportunities for detecting facility misuse. A safeguards approach for DUPIC has been developed and demonstrated, but only on a very small scale.

V. SUMMARY OF PUBLIC COMMENTS

The formal public comment period for the GNEP PEIS ended March 16, 2009. Some commenters thought the GNEP program would increase proliferation risks and some readers thought it would reduce them. Taken as a whole, the comments confirmed that all major nonproliferation issues were addressed, and the NPIA was generally recognized as comprehensive, with only a few suggestions for additional details to include. Not surprisingly, different commenters assigned different priorities to the issues addressed. Given those disagreements, and given the many hidden assumptions that may lead people to draw divergent conclusions, building a technical foundation for a stable consensus on managing proliferation risks in the nuclear fuel cycle will be challenging. There is a need for a full and open discussion of all of the nonproliferation issues, a clarification of assumptions, and careful placement into context. Starting from a recognition of the complexity of the issues, it may be possible to build a politically sustainable approach to the nuclear fuel cycle, domestically and internationally.

VI. SUMMARY

It is evident from Table 1 that the once-through and closed fuel cycle alternatives have complementary advantages and drawbacks, i.e. the strengths of one correspond to the weaknesses of the other. Once-through fuel cycle alternatives avoid production of materials that can be used in a nuclear explosive device without significant further processing, and are therefore preferable on purely technical grounds. However, this fuel cycle approach has proven challenging for the United States to implement, at least in the area of long-term sustainable spent fuel management. This seriously limits the U.S. ability to participate in offering comprehensive fuel services to other countries as a tool to limit the spread of proliferation-sensitive fuel cycle technologies. Options available for strengthening this approach are, at the front end, to expand the capacity of current providers of international enrichment services, and at the back end, to pursue international, regional, or possibly domestic arrangements for spent fuel storage or disposal.

Full actinide recycle alternatives produce materials with much higher intrinsic proliferation and security risk than a once-through fuel cycle, but offer opportunities to extend U.S. influence to address international fuel cycle challenges. The key to that influence is the ability to reduce dramatically long-term radiotoxicity hazards, which could help overcome political and public acceptance obstacles and make it possible for fuel suppliers collectively to offer comprehensive fuel services that include assured arrangements for acceptance and disposition of spent fuel. Such comprehensive fuel services offer a potentially transformative means to discourage the spread of both enrichment and reprocessing capabilities. The partial recycle alternatives may offer some of the benefits in this regard, but also suffer from most of the drawbacks of the closed fuel cycle alternatives.

Because the alternatives present complementary risks and benefits, this assessment does not identify a preferred alternative or alternatives. Such conclusions would depend on the relative importance policy makers would attach to each of the factors and the relative costs of each of the alternatives. Rather, this assessment aims to support a well-informed decision that recognizes the broad range of impacts that decision entails. Future decisions will likely involve specific technical choices for implementing the preferred alternative. For this purpose, it is important to continue efforts to develop, refine, and validate methods for proliferation risk assessment.

Going beyond current approaches to proliferation risk assessment, it would be valuable to convert the risk *assessment* methodology, which compares a specified set of technologies and processes, into a risk *reduction* methodology, which would provide design criteria to reduce proliferation risks. Such a methodology could be based on engineering and systems concepts developed to limit the known sources of proliferation risk.

Because of the importance of economic factors in determining the practicality of approaches to nuclear energy and fuel cycle systems, another important addition to existing proliferation risk assessment would be to address economic

considerations in nonproliferation analyses. Such analyses would be designed to determine where economic pressures and nonproliferation interests may be aligned, for example, in structuring fuel service arrangements to be both viable business arrangements and attractive incentives to restrain the spread of enrichment and reprocessing.

Under any of the alternatives, it will be important to continue efforts to develop effective safeguards and international security measures, for both domestic and international use. Each of the alternatives in the GNEP PEIS, including the "no action" alternative, assumes that the Department of Energy's research and development program on spent fuel recycling technologies, known as the Advanced Fuel Cycle Initiative (AFCI), would continue. This program provides critical new opportunities to develop and test advanced security measures and advanced safeguards technologies and approaches. The safeguards campaign, as part of the initiative, is focused on exploring the opportunities for improved domestic safeguards. International collaborations undertaken by the Department under AFCI could also provide expanded opportunities to develop such measures, technologies, and approaches in cooperation with other countries.

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Meeting Safeguards Challenges: Report on the 2008 INMM/ESARDA Workshop

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Abstract:

The sixth in the series of joint ESARDA-INMM workshops was held in Tokyo, Japan, on October 6 - 9, 2008, on the topic "Meeting Safeguards Challenges in an Expanding Nuclear World". Over 100 persons participated from 14 countries and 3 international/intergovernmental organizations. President Göran Dahlin represented ESARDA; President Steve Ortiz represented INMM. Keynote addresses were from Japan, the Republic of Korea, the United States, and the International Atomic Energy Agency (IAEA). Over two days, 50 presentations were given and intense discussion took place in four concurrent Working Groups. The Working Group Co-Chairs reported the results of their sessions in a closing plenary session. This paper presents a synthesis of the results of the Workshop regarding the status of international safeguards and the efforts underway to meet current challenges.

The Workshop addressed current challenges and what is being done to meet them under the topics: Safeguards implementation and State evaluation; Technical progress: The safeguards toolbox; Safeguards and nonproliferation policy and institutional issues, and Safeguards and nonproliferation academic and training programs. Key findings include: it is the State-level safeguards approach that will take us into the future; the future of safeguards will be information-driven oriented to the search for undeclared fissile material and activities without neglecting the basics of safeguards; coordinating safeguards technology development efforts continues to be important; and a mosaic of approaches have been proposed for supply assurances, proliferation resistance and other ideas, posing the question of how to knit the various bottom-up and top-down initiatives, formal and informal arrangements, legally binding and voluntary schemes together. The working group on education formulated an action plan directed at making further progress towards enhancing capacities and capabilities for education and training in the area of nuclear safeguards and non-proliferation, and proposed to form an ESARDA-INMM Working Group to pursue the ideas developed.

Keywords: safeguards; nonproliferation; ESARDA; INMM; workshop

Introduction

The sixth in the series of joint ESARDA-INMM workshops was held in Tokyo, Japan, on October 6 - 9, 2008, on the topic "*Meeting Safeguards Challenges in an Expanding Nuclear World*". Over 100 persons participated from 14 countries and 3 international/intergovernmental organizations. President Göran Dahlin represented ESARDA; President Steve Ortiz represented INMM. Keynote addresses were from Japan, the Republic of Korea, the United States, and the International Atomic Energy Agency (IAEA). Over two days, 50 presentations were given and intense discussion took place in four concurrent Working Groups. The Working Group Co-Chairs reported the results of their sessions in a closing plenary session. This paper presents a synthesis of the results of the Workshop regarding the state of international safeguards and the efforts underway to meet current challenges.

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The Opening Plenary

The Workshop got off to an excellent start with Welcome Addresses by Göran Dahlin, ESARDA President; Steve Ortiz, INMM President; and Yoshinori Meguro, INMM Japan Chapter President. Göran Dahlin, in recalling the previous five joint workshops, all of which he had attended, noted that at the 2005 workshop a special topic was included on transferring knowledge and experience to others and especially young people, and that at this workshop this topic is dealt with in Working Group 4.

Nobuhiro Muroya, Director, Safeguards Office, Science and Technology Bureau, Ministry of Education, Culture, Sports and Technology (MEXT), spoke about safeguards challenges in Japan and the efforts by Japan to meet this challenge. In order to mitigate nonproliferation concerns and prepare for the coming nuclear age, Mr Muroya proposed that the international community should strengthen the non-proliferation regime and enhance the verification ability of the IAEA. He said that the quest for 'effectiveness and efficiency' and for 'sustainability' is becoming an important issue. Therefore, it is high time to reconsider exactly what safeguards effectiveness and efficiency are and to develop indicators for improving and monitoring them. Safeguards costs could be appropriately shared with IAEA, if total resources are accurately estimated for achieving the required effectiveness. Mr Muroya said Japan is firmly committed to strengthening the three key elements of sustainable safeguards: a strong SSAC, a powerful safeguards toolbox of approaches and technology, and empowering human resources.

Hun-Gyu Lee, President, INMM Korea Chapter and President, KINAC, Republic of Korea, presented the clear and bold policy of the Republic of Korea for "low carbon dioxide and green development" to achieve a petroleum-free State that uses a low amount of fossil fuel energy. Fourteen 1000-MWe nuclear power plants are mandated for construction. To support active nuclear programs, nonproliferation and transparency are important concerns and KINAC, founded in 2006, is working to apply technology-based transparency in a comprehensive manner. With the recent initiation of integrated safeguards, Korea is working for closer cooperation with the IAEA. ROK also takes physical protection and export controls seriously and sees the necessity for regional cooperation in nuclear safety, security and safeguards.

Robert F. Cekuta, Minister Counsellor for Economic Affairs, U.S. Embassy in Japan, referred to the G8 this year committing "*to and promote the highest possible standards on nuclear non-proliferation, safeguards, safety and security (3S)".* The Global Nuclear Energy Partnership (GNEP), a grouping of 25 countries that the U.S. helped bring together, is working to address the challenges confronting countries that are maintaining, expanding, or starting nuclear power programs. GNEP's goals include advancing global energy security, encouraging clean development, and reducing the risk of proliferation. The U.S. Department of Energy has launched its Next General Safeguards Initiative aimed at revitalizing international safeguards to keep pace with safeguards challenges.

Jacques Baute, speaking for the IAEA as Director, Safeguards Information Management Division, began by mentioning the disconnect between what Member States are asking IAEA to do and the legal authority and resources made available, which will hamper effectiveness if it is not addressed. The IAEA's 20/20 Project, initiated in 2007, entailed a comprehensive review of the role of IAEA over the next one to two decades. It is projected that IAEA safeguards becomes more information driven, and up to the year 2030, in-field verification will increase 10% while evaluation effort will increase 50%. He posed the challenge for IAEA safeguards as properly balancing effort between '*monitoring the known'* and '*searching for the unknown'*.

Results from the Working Groups

Working Group 1 - Safeguards Implementation and State Evaluation Co-Chairs: Kaoru Naito, NMCC, Japan; Dieter Tillwick, NECSA, South Africa

A. Safeguards for an Expanding Nuclear World

This discussion of safeguards for an expanding nuclear world took place in the context of 50 States having announced plans to expand nuclear energy, especially in the Asian Pacific region, and an expectation of expansion in all nuclear fuel cycle activities. Statements by the IAEA Director General

at the 2008 General Conference and in the recent IAEA Safeguards Department document *"Vision 20/20"* recognize that the safeguards system needs to keep "*a step ahead*" to be credible.

As an example of this nuclear expansion, the experience with the EPR project in Finland (LWR under construction) was presented. A complex supply chain has resulted in some construction delay. Safeguards-by-design is to be incorporated. Also presented was the experience with developing a geological repository in Finland. To ensure effective safeguards, interaction with IAEA has taken place from an early stage. Geo-technical and geo-scientific information useful for safeguards is being defined and obtained. It was noted that flexibility would be needed through continuous review and development for a geological repository project lasting over 100 years.

Important examples were presented of how future safeguards measures are being developed that take account of the interconnection of effectiveness and efficiency. These included development of safeguards analytical tools to assist the European Commission (EURATOM) in near real time monitoring through process monitoring at Thorp; unattended monitoring at a PuO_2 store; and a Real Time Mass Evaluation System and Mass Slope continuous survey for gas centrifuge enrichment plants.

B. Sustainability

Looking ahead to the expansion of nuclear power generation in the Asia Pacific region, Japan has seen the need to establish the necessary infrastructure for assuring the "3S's", i.e., safeguards, safety and security. Currently, IAEA and different host countries in the region do training in safeguards. An effective mechanism for matching needs and available programs is needed, and it was suggested that a web site dedicated to this purpose would be an important tool and could be set up based on the Asia Nuclear Training and Education Program under the Forum for Nuclear Cooperation in Asia (FNCA/ANTEP[i]) model. The IAEA, or another institution like the proposed Asian Pacific Safeguards Association[ii], might take the lead in this regard.

In the U.S., a project called Next Generation Safeguards Initiative (NGSI) has been initiated by the Department of Energy, which has objective of assisting IAEA and Member States to ensure an effective international safeguards system with the capabilities to handle future nuclear expansion. It is aimed at addressing the challenges resulting from a decline in safeguards human and financial resources and the resulting negative impact on long-term safeguards technology development. The elements of NGSI are: strengthening legal authority of the international safeguards system through universalization of Comprehensive Safeguards Agreements and Additional Protocols; evolving concepts and approaches (e.g. safeguards by design and updating the 1980s Hexapartite Safeguards Project); development of technology; development of human capital; and development of safeguards infrastructure (e.g. SSAC, training and best practices).

C. Evolution of international safeguards to the State Level Approach

As the safeguards perspective of the IAEA has changed to State level from facility level, their safeguards implementation process has been modified and now involves a state-level approach (SLA), an annual safeguards implementation plan (AIP), a state safeguards evaluation report (SER), and the annual Safeguards Implementation Report (SIR).

A key finding of Working Group 1 was that it is the state-level safeguards approach that will take us into the future. While coming under integrated safeguards has become an important political milestone for many Member States, the adoption of a state-level approach is more important.

The state-level approach is characterized as being information driven, risk informed, utilizing randomization and differentiating without discriminating. Goals are to ensure consistency, enhance transparency, increase flexibility and utilize a combination of verification tools. The state-level approach is to be applied to States with all types of safeguards agreements. Implementation has been slower than expected, as effective use of available resources and obtaining adequate resources remain a continuing challenge for IAEA safeguards.

At the time of the 2008 Tokyo Workshop, IAEA had drawn the 'broader conclusion' for 48 States and integrated safeguards was being implemented in 30 States. IAEA has developed guidelines and procedures for designing a state-level safeguards approach for States under integrated safeguards. The SLA should consider State characteristics and the Agency's experience in the State. Implementation of integrated safeguards may be phased, for example, by facility type starting with light water reactors. The SLA should be subject to periodic review to take account of changes in State characteristics (e.g., expanded nuclear activities), advances in safeguards technologies and the annual safeguards conclusions drawn for the State.

As the state-level concept evolves further, the SLA for integrated safeguards should remain nondiscriminatory, and therefore a common understanding of the safeguards vision and objectives is important. Questions discussed about the SLA included:

Can it ensure a credible and cost effective safeguards system;

Can it be used to refine safeguards in States under integrated safeguards;

Can it deal with a suspicious State under integrated safeguards; and

Can it be used in a nuclear-weapon state under a Voluntary Offer Agreement?

A broader suggestion was made to consolidate all IAEA departmental involvement in a State (i.e., safeguards, technical cooperation, safety, security and assistance in nuclear technology) into a "one–house" approach yielding an IAEA-wide State specific approach and a single Country Officer for all IAEA activities. Such a change would need to be discussed with Member States.

D. State Level Approach for specific cases

In the case of South Africa, drawing the broader conclusion has been delayed due to a new nuclear energy policy being approved and then implemented, expanding nuclear activities including the pebble-bed modular reactor development, exploration and mining, the effort to quantify historical waste, and questions related to the clandestine procurement network and illicit trafficking. A road map to achieving the broader conclusion has been established.

In the case of the European Union (EU) States, specific characteristics were mentioned that could be incorporated in the SLA concept. The EU consists of 25 non-nuclear weapon States and 2 nuclear weapon States, and the EU owns all nuclear material. The standard IAEA concept is based on the State government having all the power to conceal diversion and/or undeclared activities. In the modern industrialized States of the EU, the advanced dissemination of power makes concealment difficult and there is limited potential for clandestine actions by States i.e. to divert nuclear material. It was suggested that these characteristics of the EU open the potential for considering the 25 non-nuclear weapon States as a unit for the SLA and shifting from onsite verification to monitoring through evaluation of information analysis.

E. Enhanced Cooperation with SSAC/RSAC

The IAEA has consistently identified enhancing cooperation with state and regional systems of accountancy and control as an objective. This was emphasized in the IAEA's Programme 93+2 and in its 2001 conceptual framework for integrated safeguards. Factors bearing on such enhanced cooperation were discussed. These included:

- SSACs have both national responsibilities and international responsibilities regarding application of safeguards under the safeguards agreement with IAEA;
- For enhanced cooperation with IAEA, the SSAC or RSAC must be effective; assessment of effectiveness could be based on the IAEA Guidelines and Handbook;
- The quality of an SSAC could be audited by the IAEA. Based on the results, IAEA could decide which State verification results it can use in deriving its independent findings. For this purpose the IAEA would need to establish an independent audit procedure;
- The extent of enhanced cooperation with an SSAC has been tied to having the broader conclusion drawn (How does added confidence affect IAEA's independent future findings?);
- The safeguards-relevant conclusions of an SSAC, together with accurate information and technical objectives, could be submitted to the IAEA for the purpose of confidence building and transparency; and
- The RSAC could be more fully utilized in the Agency safeguards verification. (Could enhanced cooperation with an RSAC be materialized through an added layer of control?)

The partnership concept for cooperation between the Republic of Korea and IAEA was presented. SSAC activities are categorized into enabling activities to meet State obligations; joint verification activities; and SSAC verification results to be used by IAEA in reaching its independent findings. The partnership concept is based on implementation of integrated safeguards and credible SSAC verification supplemented by IAEA quality assurance. Also included are equipment cost sharing, and allocation of responsibilities through job sharing.

Working Group 2 - Technical Progress: The safeguards toolbox
Co-Chairs: Michel Richard, CEA, France; Keith Tolk, Sandia National Laboratories, USA

The Working Group 2 discussions, involving 16 presentations, were divided into two parts: safeguards technology and equipment to support state-level safeguards and a growing nuclear world, and tools for detecting undeclared nuclear material and activities, focusing on environmental monitoring, forensics and satellite imagery.

A. Safeguards technology and equipment

Experience from safeguards implementation at large plants. IAEA experience with unattended monitoring systems at large plants has shown the importance of proper engineering of large projects. With a growing number of large facilities under safeguards, unattended monitoring systems are essential for near real time accounting in such plants.

At the Rokkasho Reprocessing Plant in Japan, important experience has been gained in developing safeguards. Discussion with the IAEA started before construction; design information examination started in 1996; design information verification started in 1999. Unattended verification systems are applied, and all safeguards data are collected by a local area network and computer system. There is an on-site Laboratory for operator and inspectorate use. The implementation of safeguards to date has shown that the initial system designs were too complex and that data acquisition is a central problem.

Developing safeguards for the Japanese MOX fuel fabrication plant is providing an opportunity to improve. A coherent, plant-wide design approach is being used. A single data acquisition method using a new universal NDA platform has been developed. And as technical issues have been identified, they are being properly addressed.

Safeguards information management and analysis. The IAEA presented the vision of future safeguards being information-driven oriented to the search for undeclared fissile material and activities without neglecting the basics of safeguards. The key for safeguards information management and analysis should be the effective *"integration of information"* from different sources. The best tool in the toolbox is the staff with their competences and skills. Therefore, it is important to bring top staff on board through outreach to staff with potential, providing training for new fields, and attracting and developing the specialists of tomorrow from universities and from within national systems. Fostering an analytical culture and strengthening IAEA's analytical capabilities are also important. Special attention should be given to the preservation of knowledge and the acquisition of the right competences.

Data and equipment authentication/security issues and Joint use of equipment. Maintaining the credibility of safeguards requires assurance that the data being collected is trustworthy. Before a safeguards finding will be fully accepted, all interested parties must be convinced that the safeguards measures used cannot be defeated. It is possible to ensure the integrity and authenticity of data and equipment against very high threat levels. To do so, the equipment and the measurement or monitoring system must have been designed with the necessary security measures in place.

A comparative study of joint-use of equipment between IAEA policy (internal Policy Paper No. 20) and the IAEA-ABACC Joint Use Equipment (JUE) agreement was presented. ABACC considers that the JUE agreement is complete and clear. It allows both agencies to make their own independent findings and observations and fulfills IAEA internal requirements.

Technologies and instrumentation. For designing the appropriate safeguards Tool Box for future nuclear fuel cycles, a key finding was that the human-computer interface matters with the bottom line being improving the efficiency of inspectors. The task is challenging due to changing nuclear power regimes and technological advances.

Synergies are possible through '*safeguards by design'* and a multi-customer approach. Sharing data from sensors with different purposes (safeguards, safety, security, personnel management) may be possible but requires addressing data, authentication concerns and cost sharing. It needs early involvement of all parties on an equal, balanced basis.

Since the equipment was designed for the last generation of bulk handling plants, new tools are available, including Monte-Carlo simulations, list mode acquisition, improved accuracy, and improved NDA measurements.

Spent fuel verification remains a challenge. New techniques for the measurement of plutonium in spent fuel are in development. Lawrence Livermore National Laboratory is currently developing a prototype, validated by experiment, which can perform partial defect verification on PWR spent fuel assemblies.

Among emerging technologies, 3-D design information verification (DIV) techniques are being successfully applied by IAEA to carry out rapid and accurate DIV in a complex facility faster and more accurately than in the past. The EC-Joint Research Centre is working with IAEA to enhance approved surveillance cameras with 3-D detection capabilities. An outdoor verification system has been successfully demonstrated to model, verify and detect modifications in facility buildings and sites. Work is progressing on the 3-D integration of other data, e.g., radiation, temperature, with applications in the security of urban areas, and on the use of 3-D modelling and verification technologies for design information verification of geological repositories

There is a need to design cost-effective and appropriate surveillance systems that enable remote inspections by means of real-time data monitoring and visualization on a spatially realistic environment and allows training of inspectors on existing safeguards systems. It would be feasible to design review software that uses information about the past to highlight similar patterns in the present data stream, which would be helpful for the inspector. Creating a 'corporate memory of safeguards-relevant events' could enable collaborative filtering and recommendations systems.

Destructive analysis and non-destructive assay. It was recommended to evaluate the complementary areas and synergies between destructive analysis (DA) and non-destructive assay (NDA) from time to time in order to improve sampling plans, sampling effectiveness and analytical efficiencies; to maximize the added value and derived information of sampling; and to reduce unnecessary manual labour.

It was acknowledged that there is a limit to achieving the safeguards goal by DA and NDA for bulk handling facilities with large throughputs (reprocessing, conversion, fabrication, enrichment) at which sigma MUF exceeds the goal quantity. Such plants require additional verification measures and inspection activities beyond routine accountancy verification measures. Enhanced containment/surveillance (C/S) measures to monitor the process should be considered.

Equipment development guidelines. Guidelines were presented for equipment developers: interact with facility planners early and often; incorporate effectiveness where possible; build in security; increase standardisation among components; build modular systems that can be upgraded individually; improve designs and component for reliability and performance; and be aware that equipment life cycles are generally shorter than facility life cycles.

B. Tools for detecting undeclared nuclear material and activities

Environmental monitoring and forensics. Environmental sampling has become a key asset for safeguards implementation and will continue to play a major role in the future. Improvements in the efficiency and effectiveness of environmental sampling remains possible. With regard to improvements in particle analysis, the use of large geometry SIMS improves significantly the timeliness and quality of the results. New methods are needed to speed up sample analysis without decreasing the quality of results while maintaining the costs.

New parameters continue to be identified. These include fissile material characterisation (e.g., age, origin) and process identification (e.g., impurities, isotopic ratios). Methods developed for Nuclear Forensic Science can be utilized in such investigative safeguards. Signatures for future nuclear fuel cycles should be investigated.

The role of the IAEA Network of Analytical Laboratories (NWAL) in supporting IAEA safeguards was stressed, as well as the central role to be played by an upgraded IAEA Safeguards Analytical Laboratory (SAL). State-of-the-art instrumentation is a necessity in future safeguards. To improve the quality of analysis, the exchange of information between the NWAL and IAEA should not be only one way.

New environmental monitoring technologies would add to the safeguards toolbox and might play an important role in the future. IAEA together with Member State Support Programs should evaluate their

benefit and the methodology for implementation. Early attention should be dedicated to the operational and legal issues (e.g., for wide-area environmental sampling).

Satellite imagery. Satellite imagery and digital image processing have become an important source of information for safeguards verification. Information is obtained from a variety of optical, thermal, hyperspectral or SAR sensors with the help of analytical tools ranging from simple to very sophisticated. Further progress is needed in several areas. Satellite-imagery-identifiable signatures (features) of materials and activities of the nuclear fuel cycle and elementary processes still need to be defined for all kind of sensors. Solutions are needed for management and easy and effective access to safeguards relevant data and information, which is the choke point of image analysis for safeguards. Fusion of different sources of satellite imagery and other sources like 3-D modelling could yield value-added to the analysis. Imagery information should be categorized and preserved.

C. Key findings of Working Group 2

From the discussion in Working Group 2, the co-chairs identified as ***basic goals*** for safeguards technology development and implementation:

- Identify the needs, i.e., the problems that need to be addressed, which are not necessarily the technologies that need to be developed;
- Identify the gaps between needs and capabilities;
- Keep the IAEA better informed about current R&D projects. Consider IAEA interest in being involved in bilateral safeguards R&D projects;
- Seek to avoid the situation where '*What we have here is a failure to communicate'*;
- Use limited human resources and funding more efficiently;
- Avoid duplication of effort; and
- Recruit more young people and encourage their creativity.

The co-chairs presented important ***observations*** coming from Working Group 2.

- One of the co-chairs expressed "*The Dream*" for safeguards technology as worldwide coordination of safeguards technology development efforts by the research and development laboratories, equipment suppliers, and the safeguards agencies, which would require mechanisms for effective communication and an environment of trust between all the participants.
- The IAEA should share with developers more efficiently their short-term, medium-term, and long-term technology development strategies.
- Industrial partners and security professionals should be brought in earlier in the development process to minimize reengineering of the system after development is 'complete'.
- Efforts should be made to share safeguards equipment with safety, security, and national safeguards authority partners when it could reduce costs and system complexity. While differing requirements and constraints make this sharing difficult, most of these problems can be solved if addressed properly.
- We must be ready to accept technology surprises, both positive and negative.
- We must be wary of declaring a problem 'solved', since conditions change and the capabilities of the adversary continue to grow.
- R&D is a continuous process. By the time a safeguards system is fully deployed, it is nearing obsolescence and so its replacement should already be in development.
- Critical reviews of R&D projects should be encouraged.
- Technology workshops, especially with attendance by people at the working level, are an effective way to share ideas.
- Setting up a Web-based information sharing system may be an effective way to share papers, presentations, and other information more efficiently. The Wikipedia model was suggested as a framework for this, but a team of volunteers to maintain the site and verify the quality of the information posted would have to be identified.

Working Group 3. Safeguards and Nonproliferation Policy and Institutional Issues

Co-Chairs: Ronald Cherry, DOE, USA; Lawrence Satkowiak, Oak Ridge National Laboratory, USA

Working Group 3 discussed a wide-range of safeguards and nonproliferation policy and institutional issues, based on a series of excellent presentations. The results are presented under the headings: nonproliferation and safeguards challenges, illicit nuclear commerce, the '3S' initiative, regional approaches, nuclear India, assurance of supply, proliferation resistance, nuclear transportation, UNSCR 1540 implementation, security culture and World Institute of Nuclear Security (WINS).

A. Nonproliferation and safeguards challenges.

One challenge of the expanding nuclear world is the additional burden on already straining nonproliferation and safeguards regimes. Moreover, the concern of nuclear terrorism from non-state or sub-state actors, the advent of clandestine networks and the diffusion and relative availability of nuclear-related information and technology have continued to compound the problem. Nuclear material remains the critical path to proliferation, which highlights the importance of safeguards and security measures.

The nature of the safeguards system and the role of the IAEA inspector have changed in response to "*shocks to the system*". The discovery of undeclared activities and facilities in Iraq was the first such wake up call. After 1991, "*completeness*" was recognized as a goal of international safeguards, along with "*correctness*". The Additional Protocol provides IAEA additional legal authority and tools to assess completeness. The use of new tools, such as environmental sampling and satellite imagery strengthened the IAEA's verification capabilities. These tools were successfully brought to bear in confirming undeclared activities in the DPRK. A presentation was made to Working Group 3 on possible scenarios for the dismantlement of DPRK nuclear facilities.

Working Group 3 addressed compliance, or noncompliance, with States obligations, how that is addressed and resolved with some special cases discussed, in particular, Iran and DPRK.

B. Illicit nuclear commerce.

The nuclear renaissance and the increase in global nuclear commerce that it will bring may create opportunities for networks to exploit gaps in regimes, but they also may create opportunities to put in place new measures and institutional arrangements to help prevent proliferation. As legitimate, peaceful nuclear commerce grows, safeguards and other arrangements need to have the ability to track illicit commerce to identify and analyze indicators of nuclear black market activities. The acquisition and analysis of information will be increasingly important for future safeguards. The Procurement Tracking System, Export Denial Information, and Voluntary Reporting Scheme are attempts to get safeguards-relevant information into the hands of IAEA analysts.

C. '3S' initiative.

An important area of international cooperation is focused on capacity building and infrastructure. The IAEA can – and does – play an important role in its international outreach and assistance programs. The "3S" (safety, security, safeguards) initiative endorsed at the G8 Summit is another, recent example of state-led undertakings to support capacity building. But knowledge management/retention issues faced by the IAEA and others can also pose a problem for capacity building efforts. In the U.S., the Next Generation Safeguards Initiative is aimed, in part, at revitalizing the technical basis, including human resources, for IAEA support and international cooperation.

D. Regional approaches.

Regional approaches to nuclear safeguards and security may supplement international and state regimes. Shared interests and shared values may make implementation on a regional level more palatable. Is it desirable to integrate those efforts into the larger safeguards and security framework? And if so, how will that be achieved?

E. Nuclear India.

A recent event highlighted in a presentation was the signature of the U.S.-India 123 Agreement by President Bush, which had followed endorsement by the Nuclear Suppliers Group (NSG). In spite of the fact that India was a non-signatory of the NPT, this was recognized as a significant event. A presentation explained India-specific safeguards and how concerns expressed in the NSG are addressed.

F. Assurance of supply.

There will be concern about assurance of supply (AOS) of nuclear fuel in an expanding nuclear world. The provision of supply assurances has long been considered without success, but some recent proposals may offer new opportunities to strengthen nonproliferation controls, for instance, through enhanced international cooperation and increased engagement with commercial players (suppliers,

vendors) who have an interest in protecting their investments. The Working Group reviewed current AOS proposals and discussed three additional variations.

Representatives from the Japan Atomic Energy Agency (JAEA) suggested an arrangement that would not impact existing fuel markets and avoid the dichotomy between "*have and have not*" states. It would include not only low enriched Uranium (LEU) fuel and enrichment services but also all elements of the front end of the fuel cycle. The proposal includes transfer of flag rights to the IAEA and the concept of "*running stock*", i.e. warehousing at existing nuclear facilities to reduce cost.

The second proposal was a non-governmental approach where consumers and commercial suppliers would share the management responsibilities for a new organization referred to as the International Nuclear Fuel Association. Its role would be to guarantee both enrichment and fuel reprocessing services (i.e., both the front and back end of the nuclear fuel cycle) to minimize proliferation risks while meeting the requirements of "*universality, transparency and economic viability*". The third proposal focused on a multi-lateral approach where enrichment facilities are owned and operated by multiple countries, and are inspected both by those countries and by the IAEA.

The government of Japan has proposed holding an international seminar on assurance of fuel supply in Vienna in order to stimulate progress in the international discussion of this issue and has suggested that both supplier and consumer states be invited. With all the proposals on the table, this seminar might be the catalyst to move forward.

G. Proliferation resistance.

Proliferation resistance is a concept that encompasses safeguards, physical protection and inherent characteristics to make it more difficult for states to proliferate but also to protect against non-state actors. Two complementary papers discussed proliferation resistance, and how applying the concept of "safeguardability" to facility designs could improve the effectiveness and efficiency of IAEA safeguards.

Proliferation resistance was defined as that characteristic of a nuclear system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices. It has both intrinsic and extrinsic features. Intrinsic features are basic technical features that could reduce the attractiveness of nuclear material and/or reduce the accessibility of nuclear facilities and could inhibit material removability, and prevent or inhibit the undeclared production of material directly usable in a nuclear weapon. It might include such things as alternate fuels or separation technologies. Extrinsic proliferation resistance measures are those that result from a State's efforts to control proliferation, such as export control, and participation in bilateral and multilateral regimes. Intrinsic proliferation resistance measures may be most valuable in facilitating the application of safeguards – the notion of safeguardability.

As the first step to promote the concept of safeguardability, the IAEA will conduct a workshop with both safeguards and facility design experts to define basic principles and fundamental design features that are critical to effective and efficient implementation of IAEA safeguards. A question raised was whether "safeguardability" will help minimize the tension between safeguards implementation and facility operations.

H. Nuclear transportation.

Two presentations addressed the transportation of nuclear and radioactive materials. Security of nuclear transportation will be impacted by the nuclear renaissance and the resulting increased global nuclear commerce as well as by the adoption of advanced fuel cycle concepts. With respect to regulations for an expanded nuclear world, the regulatory bodies in different countries are working to develop guidelines to define material categories and their appropriate protection levels. An orderly process of enhancing and expanding the laws and regulatory structures that govern safety, safeguards, and security can be foreseen, with particular attention addressed to developing countries.

I. UNSCR 1540 implementation.

The idea of coordination among safeguards, security and other systems e.g., export controls, border controls, is represented by United Nations Security Council Resolution 1540 (2004) (UNSCR 1540)[iii]. It was noted that UNSCR 1540 is not an institution in itself, but a mandate for national governments to take action. A question, however, is how to coordinate those national efforts. The 1540 Committee is not empowered to coordinate, but may facilitate coordination. State Systems of Accounting and Control can play a role in coordinating safeguards, security and other measures at the state level, but one presenter noted that the most effective approach might be to make operators and facility owners

accountable. Who should take ownership of this coordination effort? Who will be given the authority to do so and take advantage of the opportunity that UNSCR 1540 provides?

J. Security culture.

Nuclear security culture is defined in IAEA Nuclear Security Series No. 7 as "*the assembly of characteristics, attitudes and behavior of individuals, organizations and institutions which serves as a means to support and enhance nuclear security*". It is one way of "*capturing the hearts and minds of the person responsible for security*" and assists in addressing potential insider threats and strengthening the "*human factor*" in safeguards and security.

K. World Institute of Nuclear Security.

The World Institute of Nuclear Security ("WINS – Where good practices meet great ideas") was launched at the September 2008 IAEA General Conference just before the Workshop. A presentation on the intent and future plans of the WINS organization explained that WINS is targeted specifically to improve nuclear security through the promulgation of security best practices drawing upon resources both within and outside of the nuclear security community. WINS vision is of a business-centric approach that recognizes good nuclear security, much like nuclear safety, is good business practice.

L. Overview by co-chairs.

The Working Group 3 co-chairs noted that a mosaic of approaches had been proposed for addressing current safeguards and nonproliferation policy and institutional issues. The co-chairs noted that several issues discussed, including supply assurances and proliferation resistance, go back decades. They raised two general questions about the proposals made for meeting current challenges in Working Group 3. How could the various bottom-up and top-down initiatives, formal and informal arrangements, legally binding and voluntary schemes be knit together? Noting that some of the recent fundamental shifts in the safeguards system occurred in direct response to "shocks" like Iraq, will the ideas now being discussed get sufficient traction absent a similar shock today?

Working Group 4. Safeguards and Non-Proliferation Academic and Training Programs

Co-Chairs: Mark Leek, Pacific Northwest National Laboratory, USA; Willem Janssens, EC-Joint Research Centre Ispra, Italy

This Working Group exchanged experience in developing and implementing programs and curricula on nuclear safeguards, non-proliferation and nuclear security. The main topics addressed were: education and training program and curriculum development models, implementation and sustainability; content of courses, structure of programs, multi-disciplinarity; mechanisms of delivery and outreach, recruitment of students; and needs assessment for safeguards and nonproliferation education and training, and career paths.

To form a basis for the discussion, definitions and scope were addressed for: safeguards/non-proliferation, nuclear security/nuclear safety; education/training; target audience; students; professionals; specialists; generalists up to the larger public; needs identification and quantification in nuclear states and for countries planning to introduce nuclear in the future regarding future employers, career development path analysis, and potential collaboration between local, national, regional (transnational) and international activities.

A Findings of Working Group 4

1. Education and training in nuclear safeguards/non-proliferation are key elements in meeting demand that will arise as a result of the retiring workforce and the anticipated increasing demand due to the nuclear renaissance.
2. In all nuclear countries, nuclear engineers can graduate without being educated on safeguards and non-proliferation. In virtually all countries this is the rule rather than the exception.
3. As far as we can document, specialist training programs for safeguards and non-proliferation are serving current needs (US, EU, Russian Feferation). More effort is required to document the needs of government, industry and NGO's.

4. There are robust and successful programs to serve as models in education and training in both policy and technical areas (Moscow Engineering and Physics Institute, TEXAS A&M, ESARDA, Monterey)

5. The paramount need at the present time is for more active recruitment and for better utilization of education programs (by expanding capacity). Student recruitment needs to be improved significantly.

6. Outreach programs are needed to introduce and give a general base of knowledge to a broad cross section of university students and young professionals to make them aware of safeguards/non-proliferation as a career option. The same can be said about the need to increase awareness of safeguards/non-proliferation among the current nuclear workforce.

7. Outreach is needed to potential employers to bring them on board as stakeholders in safeguards/non-proliferation matters and the associated education and training.

8. There are examples of successful outreach programs for both technical and policy students. For technical students there is the U.S. Next Generation Safeguards Initiative that brings students drawn from different disciplines to the national labs as interns. For policy students in the U.S., the U.S. DOE Non Proliferation Graduate Program offers opportunities for internships at U.S.-DOE. These types of outreach programs need to be expanded (in the U.S.) and can serve as models for application in other states/regions.

9. To more accurately gauge the need for more specialists and to calibrate existing program capabilities to future needs, there must be better documentation of the current safeguards/non-proliferation workforce and its expected evolution. Such need for documentation is global and might differ strongly between technical/policy safeguards/non-proliferation compared to import/export control. Closer involvement of stakeholders (especially future employers) to specify and quantify further and additional needs is required.

10. Outreach programs are enhanced by providing opportunities for students to pursue completion of their thesis in the premises of potential future employers.

11. During professional life, the practice of rotating workforce in different safeguards/non-proliferation positions (including Junior Professional Officer's and medium-term stays at IAEA) is considered an efficient mechanism to broaden the experience.

12. Emerging nuclear countries must focus on planning and preparation of a human resource development strategy, including a needs assessment, resource assessment and implementation plan.

13. Consideration should be given to elevating the stature of safeguards/non-proliferation education programs by expanding their capacity to accommodate a national/regional constituency of students. This implies a national/regional commitment to provide the funding to expand these programs.

14. Recognition of education and training initiatives in academic programs is conditioned upon the acceptance (accreditation) of the programs by universities.

15. Success of education and training programs is determined not only by the quality of teaching staff but also by the connection to research infrastructure, typically a connection to a government laboratory or institute

16. A modular approach should be used as a way to introduce specialists in non-nuclear fields to safeguards/non-proliferation as a career option.

17. Attention should be given to lessons learned and best practices in education and training in other WMD verification regimes (e.g. chemical).

18. Education and training programs must stay abreast and on the cutting edge of latest technological developments and applications (also outside the nuclear area). This is another reason why links with research institutions are highly valuable.

19. Career paths in the safeguards/non-proliferation field are not so easily associated with regulatory requirements for education and training as is the case in nuclear safety/security. This complicates the tailoring of education and training programs to specific needs.

20. Institutions like INMM and ESARDA need to develop institutional ties to other organisations such as ANS and ENS to help their members appreciate the importance of safeguards and non-proliferation in achieving their missions.
21. An information portal (e.g. website) for use by students, teachers and interested professionals could bring significant added value in areas of job descriptions, career profiles, job opportunities, studying opportunities, reference materials, and workshops/conferences.
22. Mechanisms should be considered for capturing and transferring knowledge from experienced personnel to the next generation of safeguards specialists.
23. The initiative of the IAEA to develop guidance for a Nuclear Security Master and Certificate Program is highly welcomed and synergies with the safeguards/non-proliferation education and training should be investigated further.

B. Follow-on actions agreed by Working Group 4

Working Group 4 formulated an action plan directed to making further progress towards enhancing capacities and capabilities for education and training in the area of nuclear safeguards and non-proliferation. It was proposed to form an ESARDA-INMM Working Group to pursue the ideas developed, including common efforts to elevate the stature of education and training within INMM and ESARDA by information dissemination (via INMM and ESARDA web-sites) and seeking further INMM/ESARDA support for students.

It was proposed that this working group study and make recommendations on how to improve approaches to student outreach and recruitment; acquire greater government and industry support to build a national constituency for education programs; better document the current workforce and assess future need for safeguards/non-proliferation specialists; evaluate the synergy with other WMD threats and treaties education and training programs; promote internship programs bi-directionally (i.e. to the students and to the employers); and broaden the public to be addressed from children to diplomats. Members of Working Group 4 expressed interest in contributing to this working group.

Assessment of INMM/ESARDA Workshops and look to the future

This was the sixth in the series of joint ESARDA-INMM workshops, which were the inspiration of INMM ISD Chair Cecil Sonnier and Gotthard Stein from ESARDA. Three hosting parties were included, with rotation of the Workshops between them: ESARDA in Europe, INMM in the U.S., and the INMM Japan Chapter in Japan. They organized the first three workshops on the topic "*Science and Modern Technology for Safeguards*"[iv v vi]. They were held in Arona, Italy in October 1996; Albuquerque, U.S. in September 1998 and Tokyo, Japan in November 2000.
The second three workshops have been organized by the authors of this report, with continuity but with each workshop given a different topic and scope related to current and future international safeguards and nonproliferation, selected in consultation with senior members of the international safeguards community. These three workshops were:

- **Safeguards Perspectives for a Future Nuclear Environment,** Como/Cernobbio, Italy, October 2003
- **Changing the Safeguards Culture: Broader Perspectives and Challenges,** Santa Fe, U.S., October 2005, and
- **Meeting Safeguards Challenges in an Expanding Nuclear World,** Tokyo, Japan, October 2008.

In developing these workshops, there has been outreach to broaden participation. The Nonproliferation and Arms Control Division of INMM has become involved in organizing the workshop working group that addresses policy and institutional issues. A notable development has been the incorporation of education and training into the workshop program, providing a venue for productive discussion of this increasingly important topic. At the 2005 Workshop, a special topic was included on "*Interaction with Education and Universities*". At this 2008 Workshop, it became a full Working Group on "*Safeguards and Non-Proliferation Academic and Training Programs*", with the remarkable results reported under Working Group 4.

The success of these workshops can be attributed to the organization and structure which brought together a selected group of presenters including national and international regulators, inspectors, facility operators, developers and academics; provided ample time for presentations as the basis for discussions; and allowed two full days of intensive discussion in working groups under capable co-chairs. Reporting in a final plenary session by the working group co-chairs allowed all participants to benefit from the results of the whole workshop. Workshop proceedings have been prepared, now on CD.

Looking ahead in this series, the next ESARDA/INMM Workshop would be hosted by ESARDA and take place in 2011.

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[i] Website: www.fnca.jp/antep/

[ii] Geoffrey Shaw, *Australia's Efforts to Enhance Regional Safeguards and Nuclear Security*, Australian Safeguards and Non-Proliferation Office, June 2008, (www.jaea.go.jp/04/np/activity/2008-06-24/)

[iii] United Nations Security Council, *Non-proliferation of weapons of mass destruction*, Resolution 1540 (2004), 28 April 2004

[iv] Gotthard Stein, Stephen Dupree and Cecil Sonnier, *Results of the Joint ESARDA/INMM Workshop on Science and Modern Technology for Safeguards,* Proceedings ESARDA Conference, Montpellier (1997*)*

[v] Cecil Sonnier, Stephen Dupree, Carlo Foggi and Gotthard Stein, *Science and Modern Technology for Safeguards: Results of the Second Joint ESARDA/INMM Workshop,* Proceedings ESARDA Conference, Sevilla (1999)

[vi] Stephen Dupree, Sergio Guardini, Jim Larrimore, Cecil Sonnier and Gotthard Stein, *The ESARDA/INMM Workshops on "Science and Modern Technology for Safeguards": Looking for Technical Solutions and Directions,* Proceedings Joint ESARDA/INMM Workshop, Como (2003)

Response to IAEA "Vision 20/20" - What the SSAC could do to support IAEA to achieve its mission

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Abstract:

IAEA has prepared and published its "Vision 20/20" in February 2008. It is a strategic paper trying to answer the questions "What kind of IAEA will the world need in the timeframe up to the year 2020 and beyond?" and "How can the IAEA best fulfil that need?"

Publication of strategic "Vision 20/20" by the IAEA shows initiative and responsibility and it is therefore warmly welcomed. Member States should now in turn show similar attitude, and actively seek strategies and practical ways to help the IAEA to accomplish its extremely important "Atoms for Peace" mission. The issue of providing adequate resources was rightfully raised in the "Vision 20/20". However money or capable IAEA staff alone do not solve all the problems. This paper will propose some areas, where single Member States or a group of States can support the IAEA in its mission.

To make IAEA safeguards efficient and effective Integrated Safeguards (IS) should become universally accepted verification standard. Additional Protocol (AP) in force in a State makes it possible for the IAEA to perform more verification activities at the headquarters. This will lead to something which is called "Information Driven Safeguards". This is a learning area for both IAEA and State System of Accountancy and Control (SSAC). Open discussion and peer support between the Member States would help SSAC's to adopt the new "safeguards culture" significantly faster. Bilateral and multilateral programmes between SSAC's in this area would therefore be beneficial for the IAEA as well. This should not contradict or interfere the Agency to continue supporting States which cannot fulfil their safeguards related obligations due to lack of resources.

Integration of activities relating to safeguards, safety and security is also mentioned in the vision. In R&D it would be extremely useful to look all these areas simultaneously. Also, when applying "Information Driven Safeguards", all these areas will provide relevant information to the IAEA. Small Member States with smaller absolute resources have more relevant experience in integration of these areas and they could share their views and experience.

Member States Support Programmes for the IAEA Safeguards (MSSP) is a unique mechanism and crucial for the IAEA success. The present MSSP's should strengthen their resources and new MSSP's should become involved. Also other non conditional voluntary contributions would be valuable.

Keywords: Nuclear Safeguards Implementation, IAEA

1. Introduction

IAEA has prepared and published its "Vision 20/20" in February 2008[1]. It is a strategic paper trying to answer the questions "What kind of IAEA will the world need in the timeframe up to the year 2020 and beyond?" and "How can the IAEA best fulfil that need?" The report was prepared for Commission of Eminent persons, who provided their response by May 2008[2]

Challenges presented in the Vision and in the report of the Commission are clear. The increasing use of nuclear energy will inevitably lead to the increasing workload of the Agency. This work can not be successfully accomplished without providing additional resources to the IAEA. The Commission of

Eminent Persons proposed the IAEA budget should be doubled by the year 2020. Despite the fact, that world's economical situation has remarkably changed in a year, the big picture remains the same. Member States can not any longer expect that the IAEA can not carry out its tasks effectively.

In Nuclear Safeguards field the prospects are dim. By 2020 the number of Nuclear Weapon States may be doubled, if preventive actions are not conducted. According to Mr ElBaradei in a near future 10-20 so called "Virtual Nuclear Weapons states" may emerge, who can produce plutonium or highly enriched uranium and possess the knowhow to make warheads, but who stop just short of assembling a weapon. They would therefore remain technically compliant with the NPT while being within a couple of months of deploying and using a nuclear weapon.[3] If this phenomenon spreads, it shows NPT does not work in a way how it was intended to.

The Vision 20/20 foresees that voluntary contributions from the Member States and other entities may relief budgetary pressures to some extent. But it is not all about monetary support, the Vision 20/20 admits that the overall good cooperation with Member States are one key to success: "Even with the most sophisticated verification system, the IAEA must be able to count on the cooperation of States through State or regional systems of accounting for and control of nuclear material, systems which are required under Comprehensive Safeguards Agreements."

In the Vision 20/20 IAEA makes a few proposals where the support and cooperation of the MS's is especially appreciated:
1) Cooperation of SSAC/RSAC (mainly an implementation issue)
2) Cooperation in R&D (implementation & funding issue)
3) More information on Nuclear Trade (low level political decision + implementation problem)
4) Voluntary reporting schemes& changes in legislation (political decision)
5) Multinational Approach (MNA) fuel cycle installations (Strong MS political support needed from large group of the states)
6) Creation of new Safeguards Culture: Making Additional Protocols as a standard (Strong political support from great majority of the Member States)

This paper will focus on low level practical proposals, which can be conducted in a flexible way under the present conditions.

2. Reliable and cooperative State or Regional System

Preamble of NPT states that the signatories "Undertaking to cooperate in facilitating the application of International Atomic Energy Agency safeguards on peaceful nuclear activities NPT is based on." The cooperation is therefore a basic undertaking Member States have taken, and the lack of cooperation is therefore a breach against the Treaty. The Member States and regional Systems should take this seriously. Every SSAC (and RSAC) can, or actually should, do it homework and give this issue a good thought. Good questions for the self-assessment are, for instance:

- How the work with the IAEA could be made more efficient?
- Is the work and activities transparent to the IAEA?
- What is the quality of the declarations and other documents we are sending to the IAEA, either directly or via RRAC?
- Can we take over some activities on behalf of the IAEA, can these be agreed upon?
- Do we see pitfalls in safeguards approaches or practises being developed, which will eventually lead into heavy and costly implementation of safeguards?

Deeper cooperation with the State and the IAEA has been proposed earlier in a practical area of final disposal safeguards[4]. IAEA was encouraged to make full use of Finnish SSAC in Final disposal safeguards. The presented ideas are still valid and valuable and can be expanded to other areas. IAEA could audit SSAC's and RSAC's and based on their own decision start using their findings in their verification work.

Safety, Security and Safeguards have on common goal: to protect people from harmful effects of Nuclear and radiation related technology.[5] Therefore, all competition and contradiction of these three S-regions should be avoided. The MS's can pave the way for the IAEA by openly contacting different departments of the IAEA. Common technology solutions would be interesting. For instance FORK and

SMOPY spent fuel verification tools are used both in safeguards and safety purposes. Modern handheld radiation detection devices, on the other hand, could be utilised in all 3S applications.

4. Voluntary Support and Member State Support Programmes (MSSP's)

Required increase of the IAEA budget may not be decided in the Board and General Conference. Then, for those who are concerned about IAEA's future, there is a possibility to provide voluntary support. Actually, IAEA is becoming more and more dependent on extrabudgetary sources. This has seen to be a problem. But is it? The voluntary support given to the Agency is today under strong financial scrutiny by the auditors. There are therefore guarantees that the contributions shall not put the independency of the IAEA questionable.

The mission of Member State Support Programmes is to provide their capabilities and expertise for the benefit of Agency, where ever there is an IAEA need. Overall conclusion is that the IAEA, which lacks R&D resources and its own facilities, needs MS help to have modern state-of-the-art technology and get the training courses organised. The IAEA needs today even more MSSP support than ever, because of the lack of regular funding. The number of tasks is expected to rise.

5. Nuclear Trade Mechanisms

IAEA needs information from States, particularly with regard to procurement enquiries and export denials. To provide this information to the IAEA is bit more difficult. At least lower level political support is required, since export denials are decision of each country and confidential in nature. Moreover, IAEA is usually dealing with the SSAC, and not directly with private sector companies, which in this case are not necessarily traditional nuclear operators.

The Nuclear Trade data could be crucial for the quality of analyses IAEA is performing. Therefore, this voluntary reporting scheme would benefit global community. However, the legal justification is harder to find. IAEA has no legal access to the confidential business collaboration by private companies. And if a company would like to collaborate with the IAEA it may also be concerned about the possible economical or security risks. The safe environment must be guaranteed by the State Authorities as far as possible in to avoid any negative impact resulting from this voluntary reporting scheme.

6. Implementation of AP

Finally, the importance of Integrated Safeguards to the IAEA can not be overemphasized. It will provide substantial savings to the IAEA and at the same time increase confidence between the Member States. Every responsible state should therefore strongly urge all member states to sign and ratify Additional Protocol type agreement as soon as possible. What single SSAC can do is to inform policy makers about what is the importance of Additional Protocol on the practical level. However, one can do more with deeds than with words. Smooth implementation of Additional Protocol is the best message. Positive experiences and lessons to learned should be communicated to other SSAC's and also to the policy making level. Positive examples in implementation will encourage all to sign and ratify Additional Protocol.

7. Conclusions

Good practical level cooperation between SSAC/RSAC and the IAEA is one essential element of successful NPT regime. The only way how nuclear verification can be built upon is confidence and this can be raised through successful and practical implementation work at the ground level. Well implemented Safeguards Agreements and Additional Protocols will have an effect to the political layer and paves the way to political and financial support IAEA inevitably needs in the coming years.

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Proliferation Resistant Technical Features of Various Nuclear Energy Systems and related activities within the framework of INPRO

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Abstract:

How can different technology concepts of nuclear energy systems influence proliferation resistance? The IAEA has held several meetings on this issue since 2001. After the review and discussion by both internal and external experts of the IAEA, the results of these meetings have been documented and the document is now under the process of final review and approval. The document will be published as an IAEA NE (Nuclear Energy)-Series Report following the IAEA's final approval. The contents of the report will be reported by the IAEA staff responsible for compiling the document.

In this paper, "Proliferation Resistance" is defined as that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices.
More than 10 nuclear energy systems have been selected. Qualitative analysis of these systems, using "barriers framework" developed by the US DOE's Task Force on Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS), has been done. Proliferation resistant features and possible risks of the (examples of) selected nuclear energy systems have been identified and are reported in this paper.

On the other hand, within the framework of INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles), an assessment methodology for innovative nuclear energy systems has been developed. Proliferation resistance is included in one of the target fields of the methodology. INPRO Collaborative Project on Proliferation Resistance: Acquisition/Diversion Pathway Analysis (PRADA) was established in 2007 and is fully operational with four IAEA Member States and the European Commission. It intends to develop the appropriate methods for the identification and analysis of pathways for the acquisition of weapons usable material using an innovative nuclear system case study, and to make recommendations for evaluating multiplicity and robustness of barriers against proliferation. The current status of the project is also reported.

Keywords: proliferation resistance; INPRO

1. Introduction

How can different technology concepts of nuclear energy systems influence proliferation resistance? In expectation of a nuclear "renaissance" and for a reduction of the associated proliferation risks, over the last few years the international community has conducted substantial work on the proliferation resistance of future nuclear energy fuel cycles and related facilities.

The IAEA has held several meetings on the proliferation resistant technical features since 2001.
After the review and discussion by both internal and external experts of the IAEA, the results of the meeting have been documented and the draft of the document is now under the process of final review and approval. The document will be published as an IAEA NE Series Report. This document is to be the first report of the IAEA's survey of proliferation resistant technical

features among various technical options of nuclear energy systems.
On the other hand, within the framework of INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles), an assessment methodology for innovative nuclear energy systems has been developed. Proliferation resistance is included in one of the areas to be assessed by the methodology.

In this paper, above mentioned activities of the IAEA are introduced.
"Proliferation Resistance" is defined as that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States intent in order to acquire nuclear weapons or other nuclear explosive devices.[1]

2. Proliferation resistant technical features of various nuclear energy systems

2.1. Methodology

In the draft document, a "barrier framework" developed by the US DOE's Task Force on Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS) was used. [2], [3]
More than 10 nuclear energy systems were selected based on the following considerations:
– The system should not be only a "desk plan".
– Strong national commitment is preferable.
– The system should have "innovative" features.
The TOPS based analysis approach was applied to the selected systems in order to identify the possible impacts and tradeoffs. These impacts and tradeoffs were identified using a once through light water reactor (LWR) fuel cycle (LWR-OT) or an LWR fuel cycle using PUREX reprocessing and recycle in LWR with MOX fuel (LWR-MOX) as a reference.
This qualitative survey focuses on the intrinsic features, and not on extrinsic features or institutional aspects. It is not intended to be used for comparison or selection of a particular system, but rather to describe their technical features with respect to proliferation resistance.

2.2. Analysis of selected nuclear energy systems

The selected systems are as follows:
- Synergistic HWR/LWR
- DUPIC
- Pyro-reprocessing (IFR, DOVITA)
- Thorium Fuel Cycles
- Advanced Aqueous Reprocessing
- Small Reactors with Extended Life Cores
- IRIS
- Prismatic-fuelled HTGR
- PBMR
- Molten-Salt Reactors
- Accelerator Driven Systems
- High-Burnnp Fuel
- Spiked Fuel
- Inert Matrix Fuel

Results of the analysis of some representative examples among these systems are as follows.

2.2.1. Small reactors with extended life cores

System
There are many concepts for small, transportable autonomous reactors with long core life, no on-site refuelling, low enriched uranium fuel, highly autonomous operation and reduced maintenance. The front-end and back-end operations normally associated with the fuel cycle are presumed to be provided under strict international control**.**
Proliferation resistant features
– The long-life core designs eliminate the need to refuel the reactor, thus eliminating all 'in-country' fuel handling and storage operations.

– The system would restrict access to the reactor itself and place all fuel-cycle operations under international control. These increase the skills and access barriers relative to the reference LWR-OT case.
– Generally, long-life cores will be operated to achieve high burnups, thus increasing the isotopic and radiation barriers.
– The fact that the entire reactor could be transported fully fuelled increases the mass

and bulk barriers associated with fresh and spent-fuel transport.
– The reactor systems are designed to inhibit access (in principle, the reactor can be designed with a non-removable head, and does not require an in-containment crane or handling equipment).
– The small size of these systems may result in lower overall available masses.

Possible proliferation risks

– The fresh fuel for these concepts and variants generally use higher enrichments than standard LWRs (some approaching 20%)
  ← In contrast, the long-life, high-burnup core design means that the 239Pu content of the spent fuel is low, possibly as low as 40%.

2.2.2 Fast reactor with pyrometallurgical reprocessing (IFR, DOVITA)

System

Two main variants of pyrometallurgical reprocessing in fast reactor fuel cycles, IFR and DOVITA, use closely integrated on-site reprocessing and fuel fabrication facilities. These pyrometallurgical processes avoid separation of plutonium and complete decontamination from fission products, in particular the short half-life highly active fission products. The IFR fuel cycle uses a metal fuel and an off-line batch refuelling system, while the DOVITA fuel cycle features vibro-packed oxide fuel and uses an on-line continuous refuelling scheme.

Proliferation resistant features

– Compared with the LWR-OT fuel cycle, the IFR/DOVITA fuel cycles significantly reduce the amounts of plutonium requiring eventual disposition.
– Due to the ability to reprocess after a short cooling time, the out-of-reactor inventory is small.
– These fuel cycles can also be used to reduce existing stocks of plutonium and (depending on the specific reactor design) may be capable of doing so more efficiently than many nuclear energy systems and fuel cycles.

Possible proliferation risks (common to the fast reactors with reprocessing systems)

– Reprocessing removes some of the radiation barrier from spent fuel.
– System may aid in the spread of reprocessing technology.
– Fast reactors can produce more fissile material than they consume.

2.2.3. Sodium-cooled fast breeder reactor system with advanced aqueous reprocessing

System

The system is based on a single-cycle co-extraction process and the simplified pelletizing fuel fabrication process. It utilizes a crystallization step for the purpose of separating a major part of uranium from the dissolver solution of spent fuel.

Proliferation resistant features

– The system makes it difficult to isolate plutonium, so that plutonium with uranium and neptunium can be recovered.
– Targets of research and development activities were set to refrain from the production of separated plutonium in the fuel cycle, and to accept low decontaminated TRU fuel in order to limit its accessibility.

2.2.4. Synergistic fuel cycles utilizing HWR and LWR

System

Most heavy water reactors (HWR) employ a natural uranium based fuel cycle. Since the residual fissile content in the spent LWR fuel is higher than that in natural uranium, it can provide an excellent source of fuel for a HWR. Recovered uranium from the closed fuel cycle is an excellent source of fuel for a HWR. Examples of fuel cycle recycling technology are: DUPIC, fluoride–volatility reprocessing, and UREX+.

Proliferation resistant features

– Plutonium does not need to be separated from other transuranics and fission products.
– The fuel cycle facilities can be designed so that separation of the plutonium is difficult.
– The fuel itself is highly radioactive and extraction of the plutonium from the recycled fuel would be difficult.
– The isotopic composition of the plutonium is further degraded through additional burning in the HWR.
– The overall amount of spent fuel per MWe-hr generated is reduced.

– Intrinsic design features required to handle the high radiation fields enhance the safeguardability

2.2.5. Prismatic-fuelled HTGR

System
High-temperature gas-cooled reactor (HTGR) is cooled by gas (current designs use helium) heated to higher temperatures. In the case of the reactors considered here, the moderator (carbon) is part of the fuel design itself. HTGR can be used to burn existing stocks of plutonium.
The prismatic fuel HTGRs use a once-through, high burnup fuel using a low-fissile-density carbide fuel blocks inserted into hexagonal graphite structures arranged and handled similarly to fuel assemblies in common LWRs.

Proliferation resistant features
– The system achieves very high burnups, producing spent fuel with ^{239}Pu content on the order of only 40%, representing an increased isotopic barrier.
  ← However, it accomplishes this high burnup using low enriched uranium (LEU) fuel enriched to nearly 20%.
– Spent fuel has a slightly higher radiation barrier than spent LWR fuel.
  ← However, this may be somewhat offset by the smaller size of individual fuel assemblies implying a lower level of radiation.
– The chemical barriers associated with reprocessing spent prismatic fuel appear substantially increased relative to LWR spent fuels.
– Because of the very low plutonium density in the spent fuel, a significant fraction of the entire core must be diverted to obtain roughly a critical mass of plutonium.

2.2.6. PBMR

System
The Pebble-bed Modular Reactor (PBMR) is very similar in many aspects to the prismatic HTGR, with the main difference being that the fuel is in the form of thousands of individual tennis-ball-sized spheres that slowly move downward through the reactor, like grains of sand in an hourglass. The reactor is refuelled on-line. The pebble-bed fuelled HTGR uses a low-density carbide-based fuel using approximately 8% enriched 235U and achieves very high burnups (slightly higher than the prismatic-HTGR).

Proliferation resistant features
– Nearly all of the increments in proliferation resistance described for the prismatic-fuelled HTGR apply to the PBMR.
– A large number and large masses of pebbles would need to be handled to access a significant quantity of fissile material.

Possible proliferation risks
– The small size of the individual pebbles suggests that individual spent pebbles may be more easily shielded.
– The rapid transit through the reactor suggests that the system, particularly if modified, could provide a possible vehicle for producing weapons-useable material.
– The small pebble size, the fact that the system is continuously refuelled, and the system for sorting various pebbles (especially ‘defective’ pebbles) suggests a lower set of access barriers.

2.2.7. Accelerator driven system

System
The accelerator driven system (ADS) uses an accelerator-driven neutron source to irradiate spent fuel arising from nuclear reactors to extract additional energy from the spent fuel and to destroy the plutonium (and other actinides) remaining in the spent fuel.

Proliferation resistant features
– From a ‘proliferation resistance’ perspective, this ADS concept resembles a fast reactor preferably using pyroprocessing of the spent fuel, and has similar proliferation resistance features.
– Since the ADS eliminates fertile material from its core, it is more efficient at eliminating plutonium, therefore reducing stocks of accumulated plutonium and minor actinides (in either spent fuel or separated forms) in less time than the fast reactor systems, providing some additional overall proliferation resistance.

2.2.8. MSR

System
The Molten-Salt Reactor (MSR) can be operated either as thorium breeder within the ^{232}Th – ^{233}U fuel cycle or as actinide transmuter incinerating transuranic fuel. The system uses liquid fuel circulating in the MSR primary circuit. Main fuel processing and reprocessing technologies proposed for MSR fuel cycle are generally pyrochemical, majority of them are fluoride technologies.

Proliferation Resistant Features
– The system reduces enrichment, avoids completely separated Plutonium and reduces growing plutonium and minor actinide stocks.

Possible proliferation risks
– The MSR operated as Th-breeder requires special vigilance dedicated to ^{233}U non-proliferation.

2.2.9. Thorium fuel cycle

System
The thorium fuel cycle was originally envisaged as a closed fuel cycle, with separation and recycle of the fissile ^{233}U. There are other approaches currently being pursued that can eliminate the need to recover 233U. The thorium fuel cycles considered here are designed to use standard LWR reactors and a once-through fuel cycle system in order to optimise to burn the ^{233}U in-situ. In the so-called Radkowsky (or seed-blanket) concept, driver fuel assemblies (containing enriched uranium fuel) sustain the nuclear chain reaction and provide neutrons to blanket assemblies containing thorium, converting some of the thorium to ^{233}U that sustains the reaction. In the so-called “denatured” thorium fuel cycle, the uranium fuel and thorium are blended.

Proliferation resistant features
– The system reduces plutonium generation significantly: the thorium cycle generates about 1/3 as much plutonium as a comparable uranium-based fuel.
– The potentially higher burnups can result in degraded plutonium isotopics.
– The thorium fuel cycle could be operated with reactor grade Plutonium or ^{233}U replacing the LEU fuel. If the fuel without ^{238}U is available, the thorium cycle would generate no plutonium.

Possible proliferation risks
– The system requires the use of higher enrichments uranium (approaching 20%).
– The system produces ^{233}U, a fissile material that is useable as a nuclear explosive material.

2.2.10. Spiked fuel

System
The system uses spiking fresh fuel with radioactive nuclides to provide a radiation barrier. For example, adding the small amount of Minor Actinides (MAs) such as ^{237}Np or ^{241}Am to enhance the production of ^{238}Pu is suggested.

Proliferation resistant features
– The system increases radiation barrier.
– ^{238}Pu, which is high spontaneous fission neutron source to deteriorate the quality of the nuclear explosive and also has high decay heat to make the process of the nuclear weapon or other nuclear explosive device manufacture and maintenance technologically difficult.

Possible proliferation risks
– ^{237}Np and ^{241}Am are fissionable material and is potentially usable in a nuclear explosive device.

2.2.11. Inert matrix fuels

System
Inert matrix fuels are fuels containing no fertile material.

Proliferation resistant features
– The system avoids the generation of plutonium during irradiation, thus increases the effective rate of plutonium consumption and reduces overall plutonium stocks.
– The inert matrix is designed such that the plutonium is far more difficult to extract than that from conventional MOX fuels. Thus, these fuels can have higher chemical barriers than those of conventional MOX.

Possible proliferation risks
– Development of a technology for easily separating plutonium from the inert matrix would essentially eliminate the chemical barrier, and the higher plutonium content of the fresh fuel would represent a decreased proliferation barrier.

2.3. Consensus among the experts

There is consensuses among the participating experts that no technology would provide sufficient intrinsic proliferation resistant features alone, but this technology in combination with extrinsic measures (such as international safeguards) may help to ensure that the use of the civilian nuclear fuel cycle remains an unattractive mean to acquire material for a nuclear weapons programme. We should not say that proliferation using a proliferation resistant system is impossible; i.e., that any system is proliferation-proof.

3. Development of INPRO PR methodology

3.1. INPRO PR methodology

INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) was launched in 2000. It intends to help to ensure that nuclear energy is available in the 21st century in a sustainable manner, and seeks to bring together all interested Member States, both technology holders and technology users, to consider, jointly, actions to achieve desired innovations.
INPRO has developed methodology of assessment which can be used to evaluate whether an innovative nuclear energy system (INS) is compatible with the objective of ensuring that nuclear energy is available to contribute to meeting the energy needs in the 21st century in a sustainable manner.
Proliferation resistance is included in important areas of INPRO holistic approach. INPRO assessment methodology was published as IAEA-TECDOC. [4]
INPRO has established a set of requirements, organized in a hierarchy of basic principles, user requirements and criteria, comprising indicators and an acceptance limits in all areas that should be fulfilled by an innovative nuclear energy system to meet the overall target of sustainable energy supply.
In essence, INPRO assessment methodology in proliferation resistant area (INPRO PR methodology) can be summarised as it asks safeguards expert to confirm that nuclear facilities or system evaluated can be safeguarded effectively and efficiently.
If weak intrinsic PR levels found in the analysis, it may require more effort by safeguards. It is also to be noted that cost for safeguarding should be affordable or minimized.
INPRO defined one basic principle for PR: PR intrinsic features and extrinsic measures shall be implemented throughout the full life cycle for INS to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons program. Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself. The set of requirements in INPRO PR methodology is shown in Table 1.
INPRO methodology has been applied by several INPRO member states (Argentina, Armenia, Brazil, India, Korea, and Ukraine and so-called “Joint Study” members (Canada, China, France, India, Japan, Korea, Russia, and Ukraine)). The results of assessment study were reported and discussed at the Technical Cooperation (TC) Workshop on Lessons learned from INPRO assessment studies, held in February 2009.

At the workshop, common understanding of the INPRO PR methodology has been deepened among the participating states. Based on the feedback from the workshop it was proposed to develop a nuclear energy system assessment (NESA) support package. On this issue, the next workshop will be held in July.

3.2. PRADA

INPRO PR methodology remains to develop the methodology to evaluate User Requirements (UR) 4 regarding multiplicity and robustness of barriers against proliferation. This first requires an acquisition/diversion pathway analysis.
In 2007, the terms of reference of the project PRADA (Proliferation Resistance: Acquisition/Diversion Pathway Analysis) were agreed at the kick-off meeting and the project was launched as one of INPRO Collaborative Projects on by the following participants: Canada, China, the Republic of Korea, the USA and the European Commission.

Table 1: INPRO PR methodology

Basic Principle	User Requirements	Criteria	Indicators
BP Proliferation resistance features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme. Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself.	**UR 1 States' commitments:** States' commitments, obligations and policies regarding non-proliferation and its implementation should be adequate to fulfil international standards in the non-proliferation regime.	**CR 1.1** Legal framework **CR 1.2** Institutional structural arrangements	**IN 1.1** States' commitments, obligations and policies established? **IN 1.2** Institutional structural arrangements in support of PR have been considered accordingly?
	UR 2 Attractiveness of NM and technology: The attractiveness of nuclear material and nuclear technology in an INS for a nuclear weapons programme should be low.	**CR 2.1** Attractiveness of NM quality **CR 2.2** Attractiveness of NM quantity **CR 2.3** Attractiveness of NM form **CR 2.4** Attractiveness of nuclear technology	**IN 2.1** Nuclear Material quality **IN 2.2** Nuclear Material quantity **IN 2.3** Nuclear Material classification **IN 2.4** Nuclear technology
	UR 3 Difficulty and detectability of diversion: The diversion of nuclear material should be reasonably difficult and detectable.	**CR 3.1** Quality of measurement system **CR 3.2** C/S measures and monitoring **CR 3.3** Detectability of NM **CR 3.4** Facility process **CR 3.5** Facility design **CR 3.6** Facility misuse	**IN 3.1** Accountability **IN 3.2** Amenability **IN 3.3** Detectability of NM **IN 3.4** Difficulty to modify the process **IN 3.5** Difficulty to modify the facility design **IN 3.6** Detectability of misuse of technology or facilities
	UR 4 Multiple barriers: Innovative nuclear energy systems should incorporate multiple proliferation resistance features and measures.	**CR 4.1**: Effectiveness of features and measures **CR 4.2**: Robustness of barriers	**IN 4.1** The extent by which the INS is covered by multiple intrinsic features and extrinsic measures **IN 4.2** Robustness of barriers covering an acquisition path.
	UR 5 Optimization of design: The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized in the design/engineering phase to provide cost-efficient proliferation resistance.	**CR 5.1** Inclusion of PR in INS design **CR 5.2** Cost of PR features and measures **CR 5.3** Verification approach	**IN 5.1** PR has been taken into account as early as possible in the design and development of the INS. **IN 5.2** Cost of incorporating into an INS those intrinsic features and extrinsic measures, which are required to provide or improve proliferation resistance. **IN 5.3** Verification approach with a level of extrinsic measures agreed between State and verification authority?

The objectives of PRADA are to develop *the appropriate methods for the identification and analysis of* pathways for the acquisition of weapons usable material using an innovative nuclear system case study; and make recommendations for evaluating multiplicity and robustness of barriers against proliferation. PRADA is based on a case study on the DUPIC fuel cycle conducted by the Republic of Korea, which proposed the project. So far, two more meetings were held in May and November 2008, the next meeting will be held in the Republic of Korea in coming June.

PRADA consists of three stages:

– Stage 1: Selection of the prospective pathways;
– Stage 2: Analysis of pathways; and
– Stage 3: Assessment of multiplicity and robustness.

Discussion at the last meeting was extensively on: the threat analysis/definition (host State capabilities, objectives and strategy for proliferation), a systematic approach of pathway analysis, and the structure of the pathway analysis worksheet. Agreed systematic approach shall be used as the basic guideline for performing the PRADA analysis.

The next PRADA meeting will be held as the first meeting of Stage 2. In this stage, the first step is to collect sufficient designs and process information from the DUPIC fuel cycle system in order to perform a detailed analysis on the selected diversion pathways.

The final report is expected to be drafted in 2010.

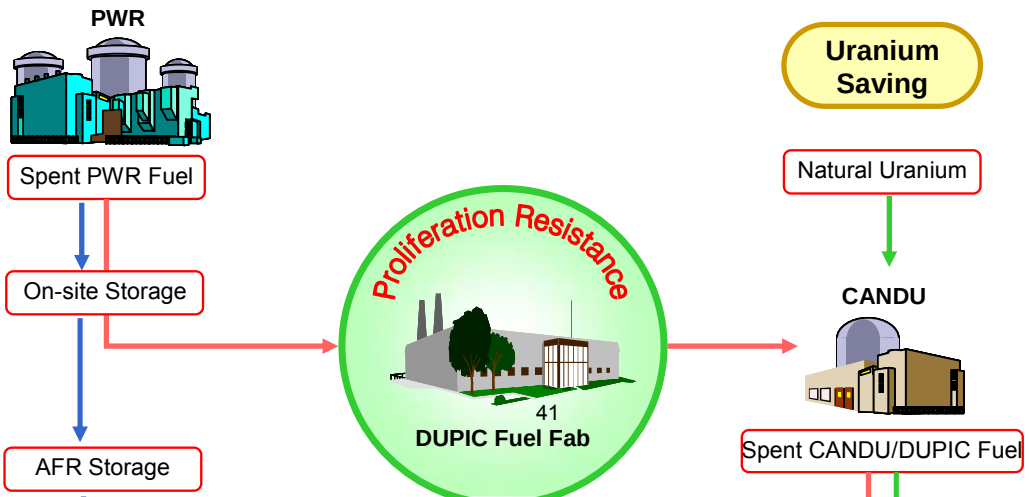

Figure 1: Concept of the DUPIC Fuel Cycle

4. Conclusion

The importance of proliferation resistance has been increased under the so called “nuclear renaissance”. Developing assessment methodology which can be used to evaluate proliferation resistance is important.
However, no nuclear energy system provides enough intrinsic features for the system to be regarded as proliferation-proof.
Qualitative survey on technical features of each system has limitation; it is not suitable for comparison of different systems.
Should we develop quantitative assessment methodology? As for some of barriers or criteria, quantitative comparison between different systems only regarding each of these barriers or criteria may be possible. However, aggregation methods to generate a single score for proliferation resistance which covers plural barriers or criteria can be misleading, possibly hiding weak links.

Pathway analysis is an important approach for developing assessment of proliferation resistance.
Some of you may have sceptical view of the importance of pathway analysis in early occasions such as early stage of development of innovative design or screening of an innovative nuclear energy system. However, proliferation resistance will be enhanced if taken into account as early as possible in the design and development of a nuclear energy system. It will be most effective if an optimal combination of intrinsic features and extrinsic measures, compatible with other design considerations, e.g. operation, safety and security, will be achieved in a nuclear energy system. From this point of view, we should take note on so called “Safeguards by Design”.

5. Acknowledgements

The survey on proliferation resistant technical features of various nuclear energy systems is work has been started in 2001 and now it is coming to the completion due to the contributors’ efforts for submitting papers as Annex part of the document, drafting and reviewing the document.
The INPRO PR methodology has been developed in cooperation with international experts and the IAEA. PRADA has been implemented also in cooperation with international experts and the IAEA. The followings are the last, but not the least of them. I appreciate all of contributors.

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"International safeguards- Basic values, institutional performance and future expectations"

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Since 1968, 40 year ago, since the signature of the Non-Proliferation Treaty, those who undertook to implement good Safeguards have been in troubles to define it. Non-Proliferation Treaty based safeguards regime, concepts and the system have passed the tests of that time period, have been proven in diverse conditions and circumstances during these dangerous years.

We must have been protected somehow from the ultimate consequences of the mutually assured destruction. We may well have been protected by the "good faith", by the spirit of the treaties and agreements we have signed. That spirit must be very strong to sustain all the bad faith and exigencies of theses past 60 years. We may well rely on it also now when the horses are again put before the cart. However, in order not to fall back to bad and dangerous practices the attention of the new horses must directed and maintained focused on the basic guiding values.

The key value examined here requires the Parties to negotiate and implement the safeguards and any disarmament measures in good faith. What does this "good" mean and imply, are again questions of fundamental importance. If we fail to "re-sign" these contracts by 2010 NPT Review Conference, we run a high risk of failing to re-establish and maintain supportive conditions for peaceful use of nuclear energy and that for the nuclear disarmament.

In order to ensure security relevance of safeguards it shall not be any more appreciated only as a political and/or technical discipline. - Thereby creating a condition that makes it possible to instrumentalize safeguards to serve also agendas that may compromise even the primary purposes. The end of the Cold War and the exigencies of the 90's and those experienced during the first years of the 21st century suggest that in order to ensure the security relevance of safeguards a re-establishment of the moral ground is needed. This implies changes in the praxis[1] at the national, regional and international levels. Changes are particularly required in institutional commitment, character qualities and attitudes, in the ways unknown situations and uncertain conditions are approached, negotiated through and measures implemented.

1. Safeguards was and still is a mission protected by *good* faith

All States and other parties, including the IAEA, are expected to cooperate in *good* faith in the implementation of the IAEA safeguards system, including national systems. There is

[1] Praxis refers here to the reality, how it works and how we are, including the consequences. This may well differ from that how we want to parade, work and how we are presenting our achievements.

nothing particular in this statement to the one well informed. In the mind of an innocent one however, the implied question comes back from the death. What does this *good* really mean and imply? How did that kind of quality, or lack of it, manifest in the past in structures, characters and performance? These are interesting questions. The latter ones however will not be elaborated within the scope of this paper.

The first 40 years of NPT-safeguards manifest for many of the involved as a heroic tail, and rightly so: Born 1968 at the height of the Cold War, survived the next 11 year period of uncertainty until 1979 and managed to establish a system and consolidate it during the following 11 years. And, then it happened 1990. - The Cold War ended, troubles started to emerge. During the following 11 years numerous known cases were addressed, some more successfully and others less. The system was also strengthened with some measures and definition of some complementary access functions. - And, then something happened again in 2001. - The faith began to erode.

The NPT-based safeguards continue to offer its services to the troubled community and its Treaty members. We may well accept that we have been protected by the *good* faith. During the years that followed however it appeared be running towards a vanishing point. Is that protection still here? -And, how about during and after the 2010 NPT-Review Conference? My answer to these questions is positive. Yes, its here, but it is not unconditional.

2. Safeguards as a confidence-building element of the NPT- regime

Safeguards are perhaps the key confidence-building element of the NPT-regime. The aim being assurance of the absence of undeclared and the absence of the diversion of declared materials and activities. Safeguards mission is hereby defined through a double negation, which means that it is genuinely a confidence building process. This does not mean that its relevance to security is diminishing, but rather contrary to that, it may well be increasing. This suggestion however implies that the very definition of security needs to be revisited later on in this paper.

The IAEA is trusted to establish reliable initial inventories, to prove the reliability not only of the information about the nuclear activities but also that of the one it is cooperating with. The IAEA is expected to continue implement its safeguards system, as negotiated with the State party. Negotiations in *good* faith are essential so that the management confidence can be established and maintained. - Thus, so that the required credible assurances are there.

One the other hand, the State (States) undertake to enable that by accepting the system and by assuming full responsibility of knowing what is happening and how it works within its jurisdiction and with other parties collaborating with it in the nuclear and related areas. States also provide the required access to relevant information and people and share the knowledge with the IAEA.

If the initial trust between the parties is not there, the conditions and circumstances are not supporting confidence- building and thereby not also supporting implementation of safeguards. In such a situation States must take *good* care that the security relevant conditions and circumstances will be appropriately addressed so as to warrant efficient implementation of the IAEA safeguards. If that is not done and safeguards implementation continues, the security relevance of all that industrious administrative effort may well be questioned. In any case, the credibility of the IAEA and its safeguards are in risk.

3. Safeguards can become directly relevant to security

Intuitively Safeguards seems to be relevant to security. When efficient it is preventing dangerous conditions to develop, preventing these to become a reality with severe consequences. And, this irrespectively how remote the danger might be. Safeguards enable us to take timely *good* note of the actual realities and to recognize the danger. Safeguards offer us ways and means to re-establish confidence and to manage attention to deficit security situations. Values such as open(mind)ness and transparency are guiding the implementation of safeguards thereby protecting us from running high risk of bringing evidence of non-objectivity and that of incompetence. Safeguards are a robust regime in the service of the community.

The contemporary safeguards are a pragmatic discipline having its merits and flaws. It makes it possible to define complex issues in functional terms. Thus, permitting and even inviting to address the complex realities in such objective terms. This makes it possible to demonstrate to the given audience, particularly to itself and other 'insiders' the administrative efficiency and excellence in implementation. Industrious effort must be invested however to ensure objectivity, correctness and consistency of the numerical representations. Also, in such environment, the performance evaluations are aimed at demonstrating the excellence of the organization. The criteria used for such evaluation are predominantly measuring administrative and management efficiency. The key issue however is about the effect, how does it feel, is it really efficient, is it *good* safeguards?

The current approach and practice provides powerful ways and means to generate reliable numbers and to establish and maintain correspondence of these with the respective hard realities. - We are strong in generating proven data and information. - Thus, we are able to take care of something important that is directly contributing to security. This necessary condition is well taken care of and must be maintained. 'Traditional safeguards', the accountancy and control discipline has direct relevance to security providing reliable facts that are proven by the trustworthy party. This is however not sufficient.

The required credible assurances imply the management confidence on institutional and organisational efficiency. - In sort, confidence on the proven reliability of all parties implementing safeguards measures, evaluating and reporting the outcomes. Safeguards services must include assurances of the absence of unknown nuclear activities and materials. - Thus, providing credible assurances of the facts and their completeness. How well is this condition satisfied, is a question for self assessment and for external audit. Are we able to implement *good* safeguards? Do we have the required resources? Are we able to address the different situations in appropriate manner and efficiently? Are the conditions and circumstances within and out of fully supporting implementation of good safeguards? Is the management interested to face this challenge? This is indeed a challenge, not technically complex nor costly, but difficult, requiring courage, open(mind)ness and readiness to accept the realities, the need to change. Consequent management of attention is required so as to let the deficit praxis retire, to let the new characters and qualities in. Safeguards-culture needs to re-volver to a confidence-building process. This requires new leadership quality!

4. Opportunity to ensure sustained relevance of safeguards

Safeguards-culture, a form of life supporting performance of the safeguards system functions in an appropriate and efficient manner. - Thus, enabling generation of proven facts and establishment and maintenance of reliable relationships as well as taking care of supportive conditions and circumstances. As a consequence, we may continue enjoying the protection by the *good* faith.

The ways we establish initial conditions and implement the safeguards measures are determining the consequences. The conditions and circumstances within and out of the organizations must also be maintained supportive. As stated above, for the generation of reliable facts we may well have all that is required, proven concepts, procedures, resources and experience. This forms the fundament. What we need now, is to manage our attention to the lack of trust and confidence, lack of 'negotiation' processes and process quality, lack of characters and character qualities. This is that what enables confidence-building. How 'good' are we in this area of responsibility?

We know that, if we do not trust the one, or we are not perceived as trustworthy, there is no number of facts, which will take care of that deficit in confidence. - Thus making consequent action impossible. Such situation is not tolerable degeneration. Some watch words could be helpful in trying to understand the nature of the problem and the implications:

- Declarations, Transparency and Openness - The key slogans of 90's. Openness, meaning open(mind)ness did disappear from rhetoric by the end of 20th century, why?
- Enforced transparency - Exerting power over the one became a prudent practice and that not always in order to achieve something of mutual interest.
- Orientation in implementation is to iron out any ambiguity, to close unresolved anomalies and to eliminate uncertainties of what ever origin.
- Protection provided by "good faith" appears to be lost, the basis for implementation is not any more a 'negotiated' one. - Trust in the NPT-Treaty based system is in risk!
- 50% of the painful conflicts we pretend to have solved during the past 60 years will revolver!
- From the mortal, societal point of view the security, as contemporarily perceived, appears to become the "chiefest" enemy of a human, like at the times of Shakespeare.

This kind of internal security-culture appears to be guided by container metaphor, which is not enabling confidence-building within or out of. The attention is managed into the organization itself rather than to the mission objectives. People in the organisation, of such a security-culture, are defensive and aggressive within and out of. As time go on this situation leads to a vicious cycle. Focus mainly on generation of objective facts creates a false feeling of security and self-confidence, inflating egos and requiring even more facts. Where this is the case, organization needs to be 'ventilated'. The responsibility for this ventilation now rests on the implementing organizations and, in case of the IAEA on its Member States. Now is the time to handle consequently.

5. Summary remarks

International safeguards are Treaty and agreement based cooperative undertaking aimed at meeting the defined objectives and serving the exclusive purposes within the given regime. To sign an agreement in *good* faith is one thing, to implement it, to comply with the letter and also with the *spirit*, is another one. In order to be able to conduct appropriately adequate resources, learning and negotiating skills as well as good guidance and supervision are much needed. Security relevance of safeguards is maintained through creation of reliable knowledge about the working conditions and implementation circumstance. The quality of this knowledge creation process is dependent on the quality of the relationships, if the reliability of the parties involved is proven or not. The Noble-Price winners have here an opportunity to provide prove of the quality so much appreciated and badly needed.

The initial establishment of the safeguards system, administrative and organization structures, strengthening measures, including provisions of the Additional Protocol and the performance of the State Evaluations by the IAEA were necessary steps in growing stronger as a system. However, there is always a risk that the measures may well be applied and tasks carried out in a manner that reduces the performance to a game or at best to a play, thus reducing, as a consequence, compliance with the Treaty and Agreement obligations to a compliance with the organization and system imperatives only. Administrative excellence and particularly the appraised performance may well provide self assurances, but does by necessity neither deter from non-compliance nor offer avenues to solve conflicting situations in a co-operative manner, as foreseen by the Treaty.

The safeguards community is resource full, particularly in respect the experiences gained. The end of the Cold War and the exigencies of the 90's and those experienced during the first years of the 21st century suggest that, to be in a position to exploit these resources efficiently, a different moral is needed to guide those who are engaged in safeguards missions.

The current period till 2012 may well be appreciated as a period for the re-establishment of the quality of that faith, a period for bringing back the meanings shrouded in the *good*. This could have something to do with the development of a security-culture the safeguards is an integral part of it. This has also something to do with the ethics of responsibility. - In short, about the moral of the 'business', about our motives and character qualities. - We, the States, the Board and the Secretariat must be playing accordingly their distinctive roles and complying with the expectations fully. - The UN Security Council, the Member States, must be taking *good* care of the dangerous situations with appropriate ways and means.

SESSION 3

DESTRUCTIVE ANALYSIS

The Evolution of a Revolutionary Concept: International Target Values for Measurement Uncertainties

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Abstract

The concept of “Target Values for Uncertainty Components” was originally conceived by the Working Group on Techniques and Standards for Destructive Analysis (WGDA) of the European Safeguards Research and Development Association (ESARDA) and matured gradually during many years. It was recognized as a valid concept for improving the reliability of safeguards measurements and for increasing the usefulness of safeguards data. It was increasingly implemented in international Safeguards as a useful tool for measurement laboratories to strive for and as a reference for Safeguards Authorities to use as a reasonable (minimum) requirement. The present paper recalls the purpose of Target Values; describes their evolution over time and provides suggestions for a review of the latest issue (International Target Values 2000, IAEA-327).

1. Introduction

Safeguarding of fissile material aims at the verification of compliance with non-proliferation treaties. Safeguards Authorities (International Atomic Energy Agency – IAEA; European Commission Directorate for Transport and Energy – DG TREN) are charged with the verification of states' or operators' fissile material accountancy declarations. This is achieved by independent measurements. The quantity of interest in these measurements is the content of fissile material in a sample taken from a bulk item at a nuclear facility. Hence, at different points in the nuclear fuel cycle, samples are taken and measured for their content in uranium or plutonium and for the isotopic composition of these elements. The results obtained by these verification measurements are then compared to the declared values. From the evaluation of such Operator-Inspector differences Safeguards Authorities draw their conclusions on the (non-)diversion of fissile material.

The accurate measurement of the fissile material content is also important to the plant operators for closing their material balance, for economical use of precious material and for most efficiently operating their facility. Consequently, both parties are interested in obtaining reliable measurement data. All this information, however, has to be achieved with limited resources in a limited period of time.

Every analytical measurement is associated with an uncertainty. This uncertainty arises from different sources, such as: sampling, weighing, conditioning, calibration and the actual sample measurement. Every step in the analytical process does contribute to the uncertainty budget of the final result.

The uncertainties in measurements of nuclear material are traditionally treated in two categories: random and systematic. In the ISO Guide for expression of uncertainty in measurement /12/, the concept of uncertainties of "Type A" and Type "B" is recommended. "Type A" evaluation of uncertainty is based on the **statistical analysis** of series of observations, whereas "Type B" is based on a method of evaluation of uncertainty by means **other than the statistical analysis** of series of observations. This concept, however, has not (yet) been considered for the issue of Target Values. Random errors typically arise from unpredicted variations, they cannot be compensated for but they can be reduced by increasing the number of observations. Systematic errors remain constant or vary in a predictable way; hence they can be compensated for (calibration). It should be noted that also calibration measurements are associated with an uncertainty. This uncertainty affects all samples in the same way /1/. It is evident that the effectiveness of verification and accountancy depends to a great extent upon the quality of the measurements achieved by both the facility operator and the safeguards inspectorate. The IAEA therefore expects that "the system of measurements on which the records used for the preparation of the reports are based, shall either conform to the latest standards or be equivalent in quality to such standards" /2/.

The purpose of accountancy and verification measurements is to enable the detection of a potential diversion of a significant amount of fissile material. This essential requirement determines the maximum tolerable Operator-Inspector difference. The significance of this difference is limited by the uncertainties associated to the respective values. Too small uncertainties, arising from too optimistic estimation or inappropriate uncertainty calculation (no error propagation, rejection of outliers), may cause artificial problems: values may appear as being significantly different. Too big uncertainty statements, due to too conservative

estimation or inappropriate uncertainty calculation, may cause problems: important differences may appear insignificant as they are “hidden” by the large uncertainties.

2. Target Values for Uncertainty Components

2.1. Historical Evolution of the Target Values

Some 30 years ago, these questions were addressed by the ESARDA Working Group on Standards and Techniques for Destructive Analysis. As a result of these discussions, the concept of "Target Values" for measurement uncertainties was created. Target Values are to be understood in this context as a requirement to the quality (actually to the uncertainty) of analytical measurements. These Values are estimates of the capabilities which can reasonably and realistically be expected from analytical laboratories. They are an essential common numeric interface between the authority requesting a certain measurement result and the laboratory executing the actual measurement. The quality of measurements varies with time as methods are improved or new methods are implemented. This has been reflected in the evolution of the concept of Target Values as well as in the actual values over the years.

Target Values are established against the background of experience based on actual performance of laboratories as proven in interlaboratory measurement evaluation programmes such as REIMEP or EQRAIN /3, 4, 5/, on experts’ advice, on Safeguards data, i.e. the analysis of Operator - Inspector differences as obtained over many years or on method performance evaluation.

The concept recognizes that the different steps in the analytical process contribute to the final uncertainty: e.g., bulk measurement, sampling, conditioning, isotope and element assay measurement. These discussions are iterative processes: procedures and values may have to be reconsidered in the light experiences gained from the various sources.

Upon initiative of the Institute for Reference Materials and Measurements (IRMM), the ESARDA WGDA pioneered the way in 1979 by presenting a list of “Target Values” for uncertainty components in destructive analytical methods to the safeguards authorities of Euratom and IAEA. These values, however, were not published. The benefits of this new concept apparently first needed to be understood and accepted by bodies concerned before going public.

The time to do so came in 1983 when revised estimates were prepared and published after four years of extensive discussions and consultations with Safeguards Authorities and plant operators /6/. With the growing international acceptance of the concept, members of two specialized committees of the INMM were involved in the next review of the Target Values. This review was published in 1986 /7/.

Only one year later the same groups undertook an attempt to consider also “random uncertainties” arising from sampling and affecting elemental assay /8/. At the time it was not possible to include values for the systematic components in sampling uncertainties.

The IAEA adopted the concept and elaborated the “1993 International Target Values” (ITV) through extensive consultation of several international and national organisations and numerous specialists. A most comprehensive document specifying Target Values for Uncertainty Components resulted /9/.

At the threshold of the 21st century, a new edition was published, the "International Target Values 2000 for Measurement Uncertainties in Safeguarding Nuclear Materials" /14/. The values were updated in the light of recent measurement experience and the presentation of the values changed significantly. Furthermore, an effort was made to bring the nomenclature in line with the latest recommendations of ISO /12/. The ITVs 2000 indeed represent target standard uncertainties.

With each revision the actual values were reconsidered and updated if this proved to be necessary and justified by experience or experimental evidence. Figures 1 and 2 give examples of the evolution of the Target Values for uranium isotope and element assay with time.

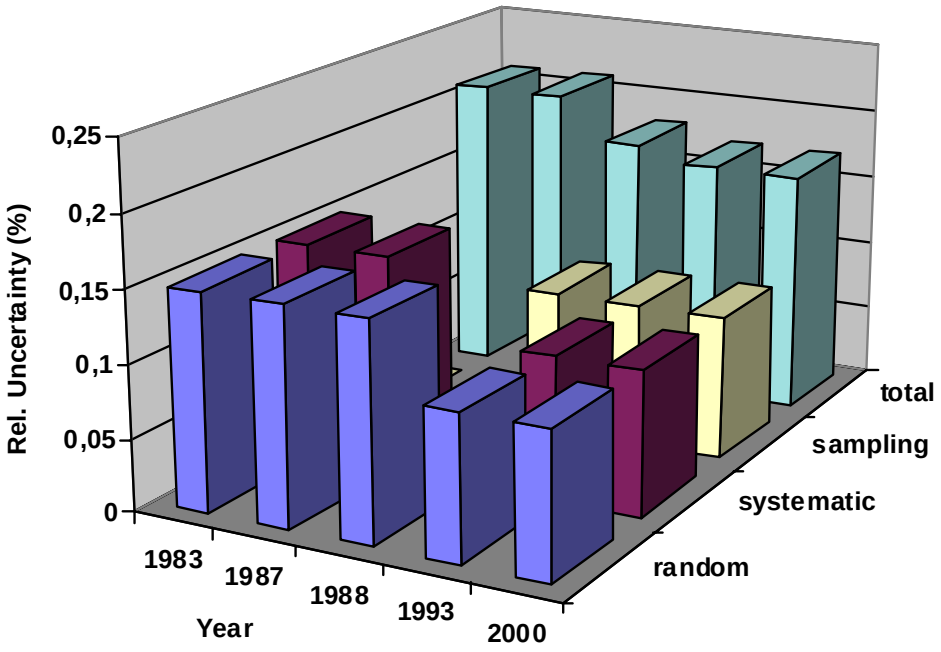

Fig. 1 Evolution of Target Values for the assay of uranium in pure uranylnitrate solution using titrimetric methods

However, not only the values have changed over the years. The first set of published Target Values /6/ focused on destructive analytical techniques, specifying desirable amounts of each material (sample) required for a particular method. Furthermore, Target Values for isotope assay were related to the abundance of the isotope of interest, i.e. consideration was given to the fact that relative uncertainties are gradually decreasing with increasing abundance (e.g. of ^{235}U).

The 1987 Target Values /7/ included K-edge densitometry, a new method which had proven to be useful for the assay of concentrated solutions of uranium and plutonium. Thus, a second so called non destructive method was considered, in addition to X-ray fluorescence. This reflected the growing tendency to use non- destructive methods as they had become more reliable, as NDA instrumentation had become more compact and as development work had led to decreased measurement uncertainties. Its application in routine analysis started to grow.

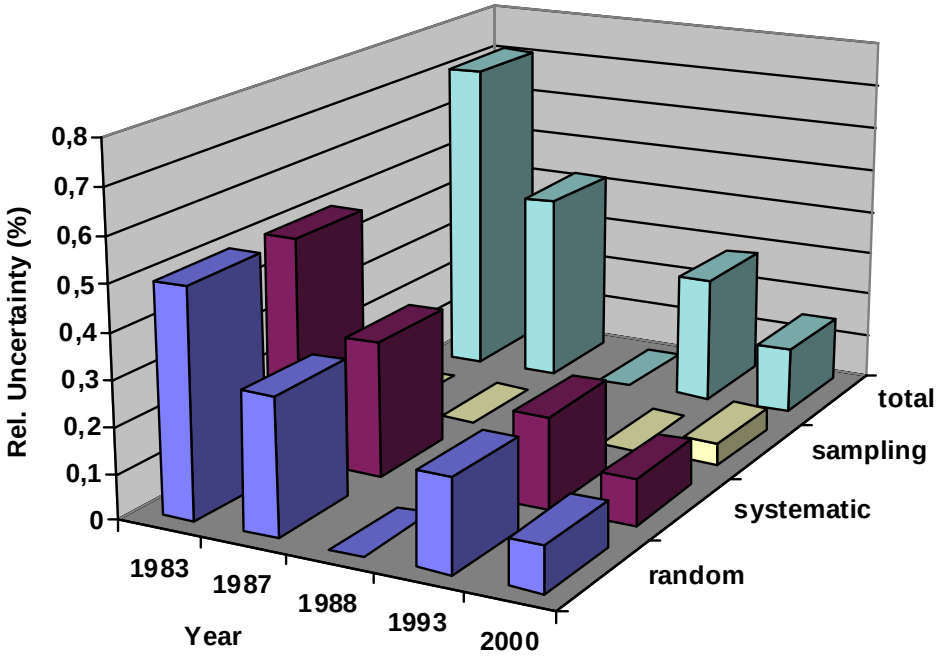

Fig. 2 Evolution of Target Values for the determination of the ^{235}U isotope abundance in low enriched uranium by thermal ionisation mass spectrometry

The 1988 release of the Target Values /8/ focused on “random uncertainties” in sampling and element assay. This was the first attempt to take into account the effect of sampling errors in nuclear material accountancy. Tables were given for various material types and different methods, giving recommendations on sample size and quoting the random measurement uncertainties already specified earlier /7/. A column stating estimates of sampling errors and another one displaying the combination of the random components of sampling error and measurement error were also given. The tables allow different sampling errors for the operator (“plant procedure”) and for the inspectors (“inspector procedure”). This arose from the fact that the number of subsamples taken and the amounts of material sampled by the operator and the inspector might be different. Additionally, effects occurring during transport from the facility to the safeguards analytical laboratory were accounted for as sampling errors. These differences in procedure might lead to different sampling errors.

In the “1993 International Target Values” /9/ the tables were presented in a far more elaborate version, structured according to material types, now including also bulk measurement uncertainties. The tables were subdivided according to the combination of methods used, yet distinguishing between uncertainty components of “random” nature and of “systematic” nature. Non-destructive methods, such as gamma ray spectrometry, neutron coincidence counting, X-ray fluorescence and K-edge absorption in their various instrumental modifications, have been fully integrated in the concept and in the tables.

Significant changes in the application of instruments and techniques took place throughout the nineteen nineties. Measurements with instruments like high level neutron coincidence counters (HLNC), K-edge X-ray absorptiometer and fluorescence analyzers (HKED) are used routinely at the plants by inspectors with great success. In consequence, the ITV 2000 issue

put more emphasis on radiometric techniques. However, significant improvements were also observed for the mass spectrometric methods, leading to a further reduction of the respective target measurement uncertainties.

2.2. Are the current Target Values achievable?

In an extensive evaluation of actually achieved measurement performance in two laboratories (the "On-Site Laboratory" –OSL- at the site of the Sellafield reprocessing plant and the "Laboratoire sur Site" –LSS- located at the La Hague reprocessing plants) were compared to the respective Target Values. The comparison is shown in table 1 for different material/method combinations. Methods include hybrid K-edge (HKED), isotope dilution mass spectrometry (IDMS), high resolution gamma spectrometry (HRGS) and thermal ionisation mass spectrometry (TIMS).

Element Assay							
Input samples	HKED (u_c(%))			IDMS (u_c(%))			
	ITV	LSS HC	OSL GB	ITV HC/GB*	LSS (LSD) HC	OSL (LSD) GB	OSL (SS) GB
U	0.25	0.25	N/A	0.28/0.18	0.11	0.07	0.10
Pu	0.67	0.65	N/A	0.28/0.18	0.12	0.06	0.10
Product samples	HKED			IDMS			
	ITV	LSS HC	OSL GB	ITV HC/GB	LSS (LSD) HC	OSL (LSD) GB	OSL (SS) GB
High U	0.25	0.25	0.20	0.28/0.18	0.11	0.07	0.14
Low Pu	0.67	0.80**	0.55	0.28/0.18	0.12	0.06	0.15
High Pu	0.25	0.55**	0.20	0.28/0.18	0.12	0.06	0.15
PuO_2 powder	0.25	N/A	0.20	0.28/0.18	N/A	0.10	0.15
(solid) MOX U	0.25	N/A	0.20	0.28/0.18	N/A	0.10	0.17
(solid) MOX Pu	0.67	N/A	0.33	0.28/0.18	N/A	0.10	0.18
Isotope Abundance Ratios (relative to ^{238}U or to ^{239}Pu, respectively)							
	HRGS			TIMS			
Uranium enrichment	ITV	LSS	OSL	ITV	LSS	OSL	
DU (< 0.3 % ^{235}U/^{238}U)	N/A	N/A	N/A	0.71	0.16	0.14	
U (0.3 % < ^{235}U/^{238}U < 1 %)	N/A	N/A	N/A	0.28	0.16	0.14	
LEU (1 % < ^{235}U/^{238}U < 20 %)	N/A	N/A	N/A	0.14	0.16	0.14	
High burn-up Pu samples	ITV	LSS	OSL	ITV	LSS	OSL	
^{238}Pu/^{239}Pu (typ. val. 0.017)	2.83	1.8	1.8	1.80	1.31	1.10	
^{240}Pu/^{239}Pu (typ. val. 0.43)	1.41	1.4	1.2	0.11	0.066	0.032	
^{241}Pu/^{239}Pu (typ. val. 0.13)	1.41	1.4	1.1	0.28	0.11	0.15	
^{242}Pu/^{239}Pu (typ. val. 0.08)	N/A	N/A	N/A	0.36	0.19	0.082	

Table 1 Routinely obtained uncertainties for different sample types, analysis methods and conditions compared to the International Target Values ITV 2000 /14/ (combined relative standard uncertainties).

(*) For hot cell (HC) and glove box (GB) conditions.

(**) The uncertainties for Pu assay are higher than the target values because no distinction is made between routine samples (with lower uncertainties) and special samples were the different matrix introduces extra error sources.

Targets are considered to only make sense if they are achievable with a reasonable effort. They should furthermore be defined in a way that they encourage laboratories to work towards an improvement of their analytical capabilities. Table 1 clearly demonstrates for the two laboratories in question that the Target Values are achievable under routine conditions. In an earlier evaluation (based on Operator - Inspector differences), the IAEA could demonstrate that Target Values are achievable in routine safeguards measurements /10/.

3. New Challenges – New Target Values ?

Nuclear material safeguards has been adapting to the changing needs and to new challenges in an evolving world. Nuclear material accountancy and its independent verification remain important pillars of modern safeguards regimes. New verification methodologies have been developed and are being implemented. These comprise also measurements of parameters in nuclear material that reach beyond the traditional nuclear material accountancy and verification.

3.1. Review of the Target Values

The concept of Target Values has been widely accepted and proven to increase the confidence in nuclear material measurements and to sustain credible nuclear safeguards conclusions. Reviewing the target values appears necessary at given intervals in order to keep pace with changing safeguards needs and improving technical measurement capabilities. The growing stocks of nuclear material and increasing number of large bulk handling facilities challenge the capabilities of Safeguards Authorities to meet their safeguards goals, especially in view of the detection of a possible diversion of a goal quantity. Figure 3 illustrates that the amount of civil plutonium has more than doubled during the period 1990 to 2003. In consequence, the inspection efforts and the uncertainty on the measurement of the amounts of material need appropriately reflect this evolution.

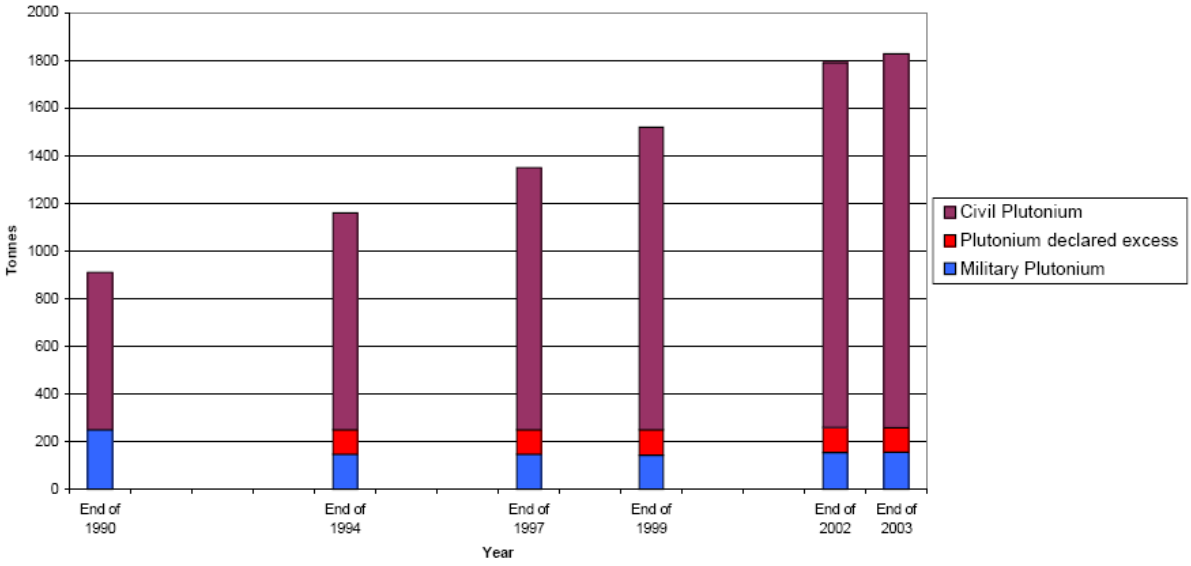

Fig. 3 Increase of amounts of plutonium between 1990 and 2003 /18/

Moreover, modern measurement equipment and improved experimental techniques enable much lower measurement uncertainties. In addition, the increasing implementation of measures of quality control and quality assurance, often in combination with the application of a proper uncertainty calculation according to the ISO Guide for expression of uncertainty in measurement (GUM), contribute significantly to improved measurement performances in the laboratory. Consequently, the International Target Values 2000 /14/ appear up for review. The IAEA has already launched a critical evaluation of their inspection data in order to propose updated values where necessary /19/. The WGDA has started discussing a review of those values relevant for destructive analysis.

3.2. Target Values Beyond Classical Accountancy and Verification

Under the strengthened safeguards regime and under the Additional Protocol, new capabilities and new tools were implemented in order to address new safeguards challenges. In particular, the consistency is checked of activities (as declared by a state or a plant operator) with measureable parameters in samples taken within a facility or in the close vicinity. Such parameters are the isotopic composition of uranium particles, the minor abundant isotopes in uranium samples (particles or bulk samples) and the chemical impurities in uranium samples. The ESARDA Working Group on Destructive Analysis (WGDA) has undertaken to analyse those tools involving (destructive) sample analysis and to discuss the associated measurement challenges, the benefits arising from measuring new parameters and the need for defining performance goals. Two dedicated workshops were organized by the WGDA and hosted by JRC-IRMM and JRC-ITU, respectively /20/. The established Target Values address measurements related to accountancy and verification. The paired comparison of operator's and inspector's measurement results is one of the main criteria for evaluation of data. In contrast to that, minor abundant uranium isotopes (i.e. ^{234}U, ^{236}U), chemical impurities and isotopic composition of uranium particles is exclusively carried out by the safeguards inspectorates. Thus, a paired comparison is not carried out and in consequence, those parameters have so far not been considered in the Target Values.

The usefulness of parameters like chemical impurities or minor uranium isotopes, however, depends strongly on the uncertainty associated to the measurement results. The group therefore strongly advocates for defining performance standards for measurement uncertainties in these areas.

4. Conclusion

The concept of Target Values for uncertainty components in the assay of fissile material was introduced more than two decades ago. Initially considered as revolutionary concept, the idea of target values has matured with increasing application and based on the experience gathered. The revisions and the new editions that were published underline the positive evolution of the concept and demonstrate the success of the idea of defining targets values for measurement uncertainties. In consequence, target values were demonstrated to be useful and internationally accepted by Safeguards Authorities and by measurement laboratories. Target Values have proven to be achievable under routine conditions.

Target Values are essentially of dynamic character. Evolving needs and improving technical possibilities favour -even demand - revisions of the values. Moreover, new parameters need to be considered in order to reflect the currently applied measurement methodologies.

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COMPUCEA 2nd generation performance evaluation

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Abstract:

COMPUCEA (Combined Procedure for Uranium Concentration and Enrichment Assay) is used for on-site analytical measurements in support of joint Euratom-IAEA inspections during physical inventory verification (PIV) campaigns in European Low-Enriched Uranium (LEU) fuel fabrication plants. The analytical technique involves the accurate determination of the uranium element content by energy-dispersive X-ray absorption edge spectrometry (L-edge densitometry) and of the ^{235}U enrichment by gamma spectrometry with a $LaBr_3$(Ce) detector. For evaluation of the $LaBr_3$ spectra a modified version of the NaIGEM code is used, which has recently been adapted to handle the presence of reprocessed uranium.
This paper describes the technique, setup and calibration procedure of the instrument. Results from PIV campaigns in 2007 and 2008 are presented, which demonstrate the performance of the technique. First results obtained with a sandwich detector configuration for enhanced detection efficiency of the passive gamma spectrometry are discussed.

Keywords: uranium elemental analysis; uranium enrichment; X-ray absorption edge spectrometry; lanthanum bromide detector; NaIGEM analysis code

1. Introduction

The Combined Procedure for Uranium Concentration and Enrichment Assay (COMPUCEA) represents a testing method for the uranium element and ^{235}U-enrichment assay routinely applied to the analysis of uranium product materials (uranium oxide powders and sintered uranium oxide pellets). The actual analyses on this type of sample materials are performed with mobile equipment in different European fuel fabrication plants for Low-Enriched Uranium (LEU) fuels during the nuclear material accountancy verification activities of international nuclear safeguards authorities (Euratom, IAEA). The analytical support provided on site by analysts from the ITU during the PIV campaigns involves the accurate determination of the uranium element content and of the ^{235}U enrichment in verification samples selected by the Safeguards inspectors according to a defined sampling plan.

2. Analytical procedure

The general scheme of analysis followed in the measurements with COMPUCEA [1] is outlined in Figure 1. It includes the following 3 major steps:

1. *Sample preparation:* The purpose of this first step is to transform the solid uranium samples (powders or pellets) into a nitric acid solution of approximately constant acidity (3 M) and uranium concentration level, and then to carefully characterise the obtained solution for its density and temperature. The nominal uranium concentration is set to be around 190 mgU/ml, which is close to the upper limit of the linearity range of the new L-edge densitometry measurement [1].

2. *Radiometric measurements (L-Edge Densitometry and Gamma Spectrometry):* Aliquots are taken from the sample solution and subjected, without any further treatment, to parallel L-edge densitometry and passive gamma counting measurements. The two radiometric measurements are described in more detail below.

3. *Data evaluation:* In the final step of data evaluation, the different pieces of information obtained from the sample preparation and from the two radiometric measurements are combined to evaluate the uranium weight fraction in the original sample and the ^{235}U weight fraction in the uranium material. It should be noted that the two radiometric measurements are interdependent, i.e. each technique requires input from the other for final data evaluation: the L-edge densitometry measurement needs the knowledge of the enrichment for the calculation of the uranium atomic weight, and the gamma measurement needs as input the knowledge of the uranium concentration. The evaluation of the final uranium concentration and enrichment is therefore made in an iterative manner.

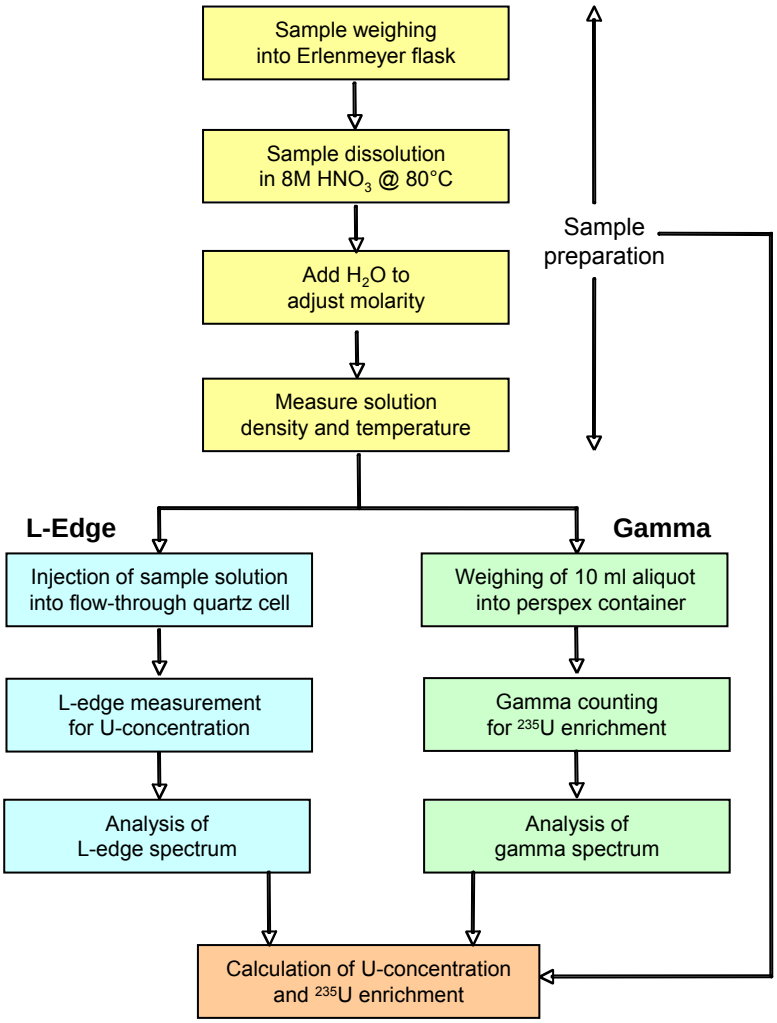

***Figure 1**: General scheme of analysis for COMPUCEA.*

2.2. Uranium elemental assay

In the 2nd generation of the COMPUCEA equipment, a miniaturized 30 kV/100 µA X-ray generator, and a Peltier-cooled, high-resolution 10 mm^2 x 0.5 mm Si drift detector are used to measure the energy-differential absorption at the L_{III}-shell absorption edge, which occurs for uranium at the energy of 17.17 keV. A representative measurement example together with a cross-sectional view of the equipment is shown in Fig. 2. In this setup, the sample cell consists of a fixed flow-through quartz cell with a path length of 2 mm, and a cell volume of 125 µl.

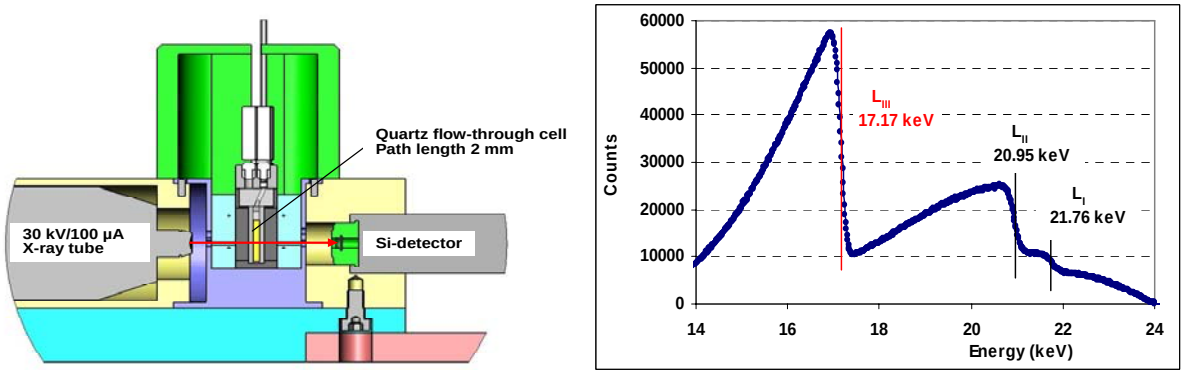

Figure 2: L-edge absorptiometry with an X-ray continuum used in COMPUCEA 2nd generation. Left: Measurement setup. Right: Measurement example.

The evaluation of the uranium concentration from the measured ratio of photon transmission across the L_{III} edge at 17.17 keV follows the proven analysis procedure adopted for K-edge densitometry with an X-ray continuum [2]. In this approach, the photon transmission as a function of energy, T(E), is measured relative to a blank spectrum from a nitric acid solution of representative molarity (3M), and then linearized in a representation lnln(1/T) *vs* lnE. Linear least-squares fits to the respective data on both sides of the absorption edge determine the photon transmission at energies slightly displaced from the absorption edge ('non-extrapolated fitting mode', E_+ = 17.60 keV, E_- = 16.70 keV), or directly at the absorption edge energy ('extrapolated fitting mode'). Fitting intervals ranging from 15.50-16.70 keV, and from 17.60-18.80 keV were chosen for the evaluation of the transmission ratio across the L_{III} edge.

In the 'extrapolated fitting mode', where the photon transmissions are determined directly at the absorption edge energy, the uranium volume concentration ρ_U (in g/cm^3) in the measured solution is obtained by the following relation:

$$\rho_U = CF_{AW} \cdot \left[\frac{\ln\left[T(E_-)/T(E_+)\right]}{\left|\Delta\mu_U\right| \cdot D} \right] \quad . \qquad (1)$$

Here, the quantity $|\Delta\mu_U|$ (in cm^2/g) describe the difference in photon mass attenuation coefficients of uranium at the energies E_- and E_+ and D (in cm) denotes the path length of the photon beam through the sample cell. The term CF_{AW} is a correction factor accounting for the atomic weight of the uranium under analysis. It is calculated from the known enrichment. The uranium concentration derived in the 'extrapolated fitting mode' from the transmission ratio directly at the L-edge energy is virtually insensitive to matrix effects.

However, the availability of two independent analysis results for the uranium concentration from the extrapolated and non-extrapolated fitting analysis represents a very useful diagnostic tool. A statistically significant difference observed between the two results will immediately point to any sort of deviation in the matrix composition of the measurement sample from the assumed 3M HNO_3 reference

matrix. This knowledge is not of immediate relevance for the L-edge densitometry measurement, but of practical help for the parallel enrichment measurement, where any deviation in the matrix composition will have a direct influence on the gamma attenuation behaviour of the sample.

A practical example refers to the analysis of uranium samples containing a significant amount of gadolinium. In this case the difference observed between the uranium results from the non-extrapolated and extrapolated fitting analysis can be used for an estimate of the Gd content, provided the discordance between the two evaluated uranium results can be reasonably attributed to the presence of this additional element alone. The knowledge about the Gd content then allows calculating corresponding correction factors for the enrichment measurement.

2.3. ^{235}U enrichment determination

The ^{235}U enrichment measurement in COMPUCEA is based on the counting of the ^{235}U 186 keV gammas of a defined amount of uranium in solution in a well-defined counting geometry. The new detector replacing the previous HPGe well detector is a standard-type 2" x 1" cerium-doped lanthanum bromide scintillation detector – $LaBr_3$(Ce). It offers the main advantage of being a detector operating at room-temperature, therefore eliminating the need for detector cooling with liquid nitrogen as required before. This practical advantage, particularly for in-field applications, largely compensates for the drawback of an inferior energy resolution (FWHM @ 186 keV about 9 keV for the LaBr detector compared to a value of 1.3 keV obtained with the previous HPGe well detector). Fortunately, the relatively simple gamma spectrum of ^{235}U allows accurate enrichment measurements also at this lower degree of energy resolution [3].

Since the recently developed $LaBr_3$(Ce) scintillation detectors are not yet available in the form of well-type detectors, the configuration of sample counting was changed to a counting geometry with the sample located on top of the detector. As this measurement geometry is less efficient than counting in a well detector, the sample volume for the measurement with the $LaBr_3$(Ce) detector was increased from 2.5 ml (as used before in a HPGe well detector) to 10 ml.

The proper evaluation of the measured gamma spectrum for an accurate enrichment determination involves a two-step process: (1) analysis of the gamma spectrum itself for the extraction of the 185.7 keV net peak counts, and (2) the calculation of appropriate correction factors for the extracted peak counts accounting for the impact of variable sample parameters. In the first step, a modified analysis code based on the NaIGEM code previously developed for the analysis of uranium gamma spectra measured with NaI(Tl) scintillation detectors [4] is used for the deconvolution of the gamma spectra measured with the new $LaBr_3$ scintillation detector. Examples for fitting graphs are shown in Figure 3.

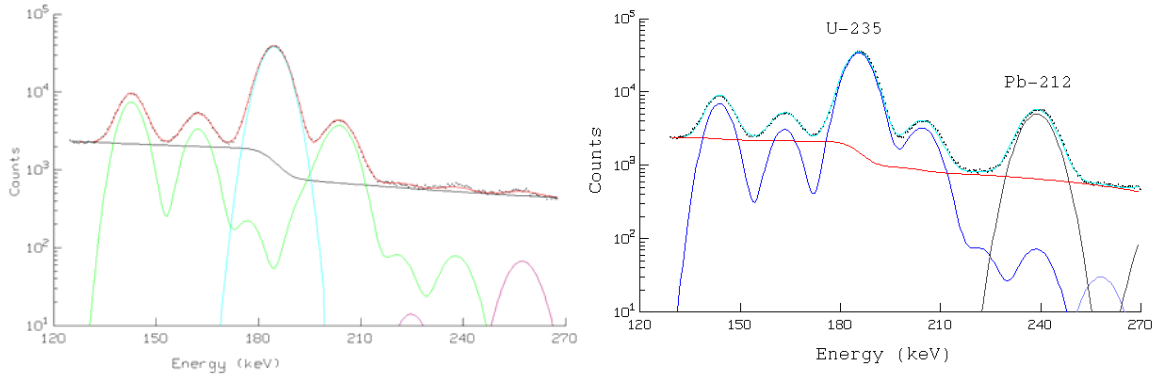

Figure 3: Examples of fitting graphs for $LaBr_3$ gamma spectra taken from low-enriched virgin uranium (left) and recycled uranium (right) samples.

The software for spectrum analysis determines the net peak counts for the most prominent ^{235}U gamma line at 185.7 keV with associated uncertainty. Additional information is provided on the quality of the response function fitting, and on the full width at half maximum value determined for the 185.7 keV line. The fitting code has recently been adapted to handle gamma spectra from recycled uranium materials which show an additional gamma ray at 238.6 keV originating from the ^{232}U descendent

^{212}Pb (see example on the right side in Fig. 3). For those spectra, the analysis code also reports the peak area with associated uncertainty for the 238.6 keV line.

In the second step of the analysis, correction factors for the evaluated 185.7 keV net peak counts are calculated from known sample parameters. The major part of this calculation work is accomplished by means of a tailored Monte Carlo simulation of the gamma detection process. The simulation software is able to calculate, on a relative basis, for the modelled measurement configuration the detection rates for the 185.7 keV photons in dependence of crucial sample parameters. The relative detection rates calculated within a runtime of 100 s for the Monte Carlo calculation have a statistical precision of ca. 0.03%.

The sample parameters taken into account in the Monte Carlo calculation include:

- the uranium concentration,
- the gadolinium concentration (if this element has been detected),
- the solution density (representing also a measure for the HNO_3 molarity),
- the bottom thickness of the sample container, and
- the sample volume.

The relative detection rate delivered by the Monte Carlo simulation program is normalised to the detection rate calculated for a reference sample, yielding a final correction factor *CF(U,Gd,ρ,Bot,V)* for the measured 185.7 keV net peak rate.

Another correction factor, *CF(Pa)*, accounting for a small contribution of interfering Pa gamma rays to the 185.7 keV line, is calculated separately. The ^{238}U daughter products ^{234}Pa and ^{234m}Pa, which normally are in secular equilibrium with ^{238}U, emit weak gamma rays with energies at 186.15 and 184.7 keV, which are close to the main ^{235}U gamma ray at 185.72 keV and cannot be resolved from the ^{235}U gamma ray. Their contribution to the observed peak intensity at 186 keV, though very small, should be not ignored in high-accuracy enrichment measurements. The numerical value for *CF(Pa)* calculates from the respective photon emission probabilities to:

$$CF(Pa) = 1 - \frac{(100 - enr)}{enr} \cdot 1.2223 \cdot 10^{-5} \qquad (2)$$

where *enr* denotes the ^{235}U enrichment in wt%. The correction can be only applied when the enrichment is approximately known. The numerical factor in Eq. 2 represents the ratio of the emission rates (^{234}Pa+^{234m}Pa)/^{235}U.

3. COMPUCEA calibration

Both measurement techniques in COMPUCEA require an instrument calibration. The calibration approach has been revised and simplified in the sense that for each technique in principle only a *single calibration factor* needs to be determined. In order to arrive at this favourable situation, measurement and instrument properties considered as being relevant for measurement performance and calibration have been carefully studied during the instrument development stage.

Prior to the in-field measurements, all COMPUCEA systems are calibrated at ITU with a set of suitable reference solutions. With this pre-calibration, combined with the quantitatively known correction factors to be applied, calibration in field is reduced to the measurement of two calibration samples for a verification or re-normalisation of the basic calibration factors determined at ITU.

3. 1. Reference materials

The reference materials available for calibration consist of a set of sintered UO_2 pellets with 3 different enrichment grades (0.72, 2.10 and 4.40 wt% ^{235}U). The UO_2 pellets were previously taken from the production batches of a uranium fuel fabrication plant, and then subsequently characterized by primary analytical methods for the uranium element content and isotopic composition. The analytical measurements for material characterisation were independently carried out by ITU and IAEA-SAL.

The reference solutions required for instrument calibration, either in field or at ITU, are prepared from these reference pellets, following exactly the same procedures for sample preparation as applied for the normal measurement samples. From each reference material, a single reference solution is prepared, which is used both for the calibration of the L-edge densitometer and of the gamma spectrometer.

3. 2. Calibration of the L-edge densitometer

For calibration with the pure uranium reference solutions the densitometry equation (1) is applied. The calibration factor to be determined from the measured transmission ratio $T(E_-)/T(E_+)$ is the value of $\Delta\mu_U$:

$$\left|\Delta\mu_U\right| = CF_{AW} \cdot \left[\frac{\ln\left[T(E_-)/T(E_+)\right]}{\rho_U \cdot D}\right] \tag{3}$$

The factor CF_{AW} accounts for the uranium atomic weight. For the calibration solutions, the quantities CF_{AW} and ρ_U are known reference values. The path length D of the measurement cell is also known (D = 0.2 cm), but specified from the manufacturer with a relative uncertainty of 0.5%. This uncertainty has no influence on the overall measurement uncertainty as the same cell is used for the calibration and all measurements. For the evaluation of $\Delta\mu_U$ the nominal value D = 0.2 cm is used. Any deviation of the true path length from this nominal value is then accounted for in the determined value of $\Delta\mu_U$.

3.3. Calibration of the gamma spectrometer

The main task of this calibration is to establish a proportionality or calibration factor, *K*, relating the amount of ^{235}U in the measurement sample, $M(^{235}U)$, to the measured and evaluated net peak counts in the 186 keV line from ^{235}U, P_{186}. For the calibration (and for the sample measurements later as well) the correction factors $CF(U,Gd,\rho,Bot,V)$ and $CF(Pa)$ mentioned in Section 2.3. have to be taken into account, leading then to the following calibration expression:

$$M(^{235}U) = K \cdot P_{186} \cdot CF(U,Gd,\rho,Bot,V) \cdot CF(Pa) \tag{4}$$

The determination of the basic calibration factor, *K*, is made for a defined measurement condition, characterized by the following reference values:

Uranium concentration:	190.00 g/l
Gadolinium concentration:	0.00 g/l
HNO_3 acidity:	3M
Sample volume:	10.00 ml
Bottom thickness of sample container:	1.10 mm

The corresponding measurement parameters are typically kept close to these reference values, both for the calibration and for the routine measurement samples (except for the Gd content in the routine samples, which is taken as found). The correction factor $CF(U,Gd,\rho,Bot,V)$ applied to the measured 186 keV peak count rate per g ^{235}U, then just represents the ratio of the detection rates calculated for the actual sample and for the above reference conditions. The applied corrections are typically far less than 1%. With the adopted calibration procedure the calibration for the enrichment measurement therefore simply reduces to the determination of a single calibration factor *K*.

4. COMPUCEA performance evaluation and validation

To evaluate the performance of the COMPUCEA 2nd generation equipment, the influence of relevant measurement parameters, such as working and linear range, matrix effects, counting precision, measurement reproducibility, gamma self-attenuation and counting geometry, was studied in detail [5]. With the identification and quantification of individual uncertainty components, it was then possible to present an estimate of the total uncertainty of the two analytical determinations made. This estimation also includes uncertainty components related to the sample preparation (sample weighing, dissolution and density measurement).

The estimations for the total measurement uncertainties of COMPUCEA 2nd generation are well within the International Target Values (ITV) for measurement uncertainties in the field of International Safeguards for nuclear materials [6], as listed in Table 1. The typical counting times with the 2nd generation of COMPUCEA are 3 x 2000s, however, even with a reduced measurement time of 1000 s, the ITV's are safely met.

Analysis	Total uncertainty (% rel. Std. Uncertainty)		
	2nd COMPUCEA (3 x 2000 s)	2nd COMPUCEA (1 x 1000 s)	ITV (1000 s)
U-concentration	0.13	0.20	0.25
^{235}U abundance	0.26 a)	0.40 a)	0.45

a) For a medium enrichment of 2 wt% ^{235}U.

Table 1: Performance data for the 2nd generation of COMPUCEA.

For method validation, the measurement performance was then evaluated in three different ways by comparing the COMPUCEA results

i. with results from parallel analyses made with a primary reference method,
ii. with well-specified reference values for the quantity of interest, and
iii. with data obtained in round robin tests

4.1. Validation of U-concentration measurements

During the in-field measurements made with the 2nd generation of COMPUCEA in 2007 and 2008, a total of 115 uranium samples were analysed at 4 different locations. For a subset (taken at 3 of the 4 locations), parallel samples were taken for remote analysis with a qualified primary analytical method (potentiometric titration according to the method of Davies and Gray). The measurement uncertainties (1s) for the reference method were specified to be 0.05% for the random error, and 0.05% for the systematic error.

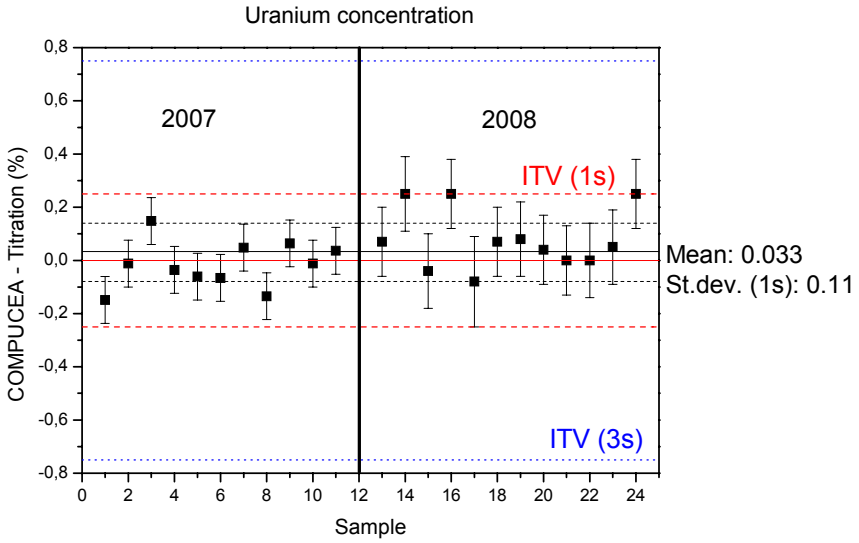

Figure 4: Plot of relative percentage differences between COMPUCEA L-edge in-field analytical results and the primary reference method titration.

Figure 4 shows the results of the parallel analyses. The COMPUCEA results are in perfect agreement with the titration results (average difference of 0.033%). The combined random uncertainty of both methods (0.11% for COMPUCEA and 0.05% for titration) calculates to 0.12%. The observed standard deviation of 0.11% for the differences between the analysis results is in agreement with this value.

The second approach for measurement performance evaluation bases on a comparison of the COMPUCEA analysis results with tightly specified reference values for the quantity of interest. This comparison applies to the uranium content in sintered UO_2 fuel pellets manufactured as reactor fuel. The very tight specifications for the oxygen-to-metal ratio (2.00±0.01) and for element impurities restrict the uranium content in this kind of nuclear materials to the very narrow range of 88.11-88.16 wt%. This margin holds for all pure uranium fuels produced around the world [7]. For the mean value of 88.135±0.025 wt% this uncertainty range means a relative uncertainty of ±0.028%.

Among the total number of 57 samples analysed in 2007 there existed 29 sintered UO_2 pellet samples with this kind of well-specified uranium content. Figure 5 displays the relative percentage differences between the COMPUCEA analysis results and the specified reference value for this set of samples. The paired data show an average difference of -0.015 % with a relative standard deviation of 0.08 %. Both values are well within the ITV-values (0.015% for systematic, 0.2% for random effects) for the COMPUCEA measurement uncertainty.

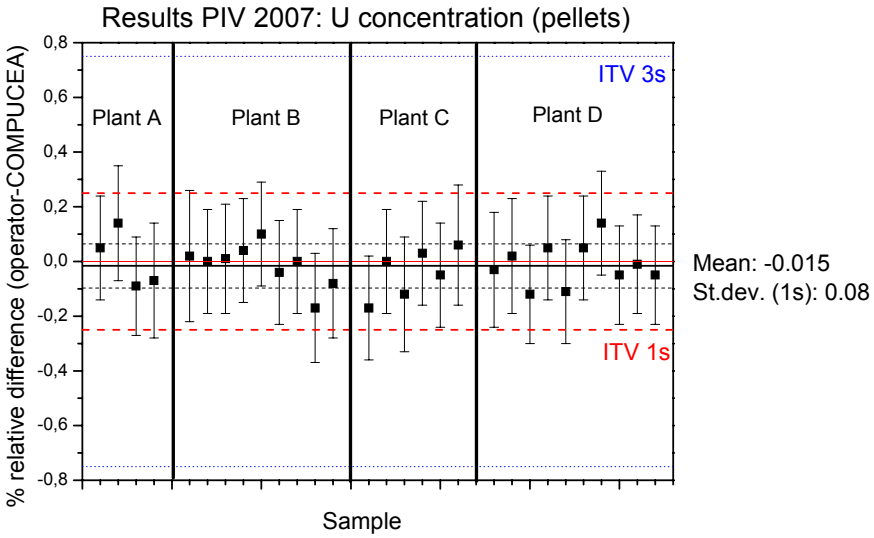

Figure 5: Plot of relative percentage differences between COMPUCEA analysis result and specified reference value for 29 UO_2 pellet samples.

	Reference values (g/kg)	COMPUCEA result (g/kg)	% relative difference (COMPUCEA-reference)
EQRAIN 12 1st round	210.81 ± 0.11	210.75 ± 0.15% 210.73 ± 0.15%	-0.028 -0.038
EQRAIN 12 2nd round	221.40 ± 0.11	221.38 ± 0.15% 221.37 ± 0.15% 221.33 ± 0.15%	-0.009 -0.014 -0.032
EQRAIN 12 3rd round	238.14 ± 0.11	238.09 ± 0.15% 238.06 ± 0.15% 238.00 ± 0.15%	-0.021 -0.034 -0.059
EQRAIN 12 4th round	249.70 ± 0.25	249.79 ± 0.15% 249.65 ± 0.15% 249.63 ± 0.15%	0.036 -0.022 -0.029

Table 1. COMPUCEA results obtained under EQRAIN 12. The uncertainties stated are 1s.

A third key element for measurement performance evaluation, and according to the Laboratory Guide to Method Validation [8] the preferred way of validating methods, is the external control through interlaboratory trials. An interlaboratory round robin exercise pertinent to the validation of COMPUCEA is provided by the programme "EQRAIN" (from the French acronym for "Quality Assessment of Analysis Results in the Nuclear Industry") conducted by CEA-CETAMA. Under the EQRAIN programme highly concentrated uranyl nitrate solutions (> 200 gU/kg) of unknown concentration are distributed on a regular basis to interested parties for controlling their analytical methods for uranium analysis. COMPUCEA is part of this programme for measurement control. The results from 4 round robin tests performed under EQRAIN 12 in 2008/2009 are summarized in Table 1. During each of the rounds, 2 or 3 sub-samples were analysed independently by COMPUCEA. The results obtained prove a high degree of reproducibility and, from a comparison with the reference values communicated after submission of the results, a high degree of accuracy with an average deviation as low as -0.023%.

4.2. Validation of ^{235}U enrichment measurements

The validation of the enrichment measurements made with the new $LaBr_3$ detector setup bases on a comparison with results obtained with a recognized and validated measurement technique for isotope abundance measurements, i.e. Thermal Ionization Mass Spectrometry (TIMS). During the in-field COMPUCEA measurement campaigns in 2007 and 2008 a total of 17 parallel samples of low-enriched uranium were taken and sent for off-site analysis by TIMS.

The compared data are plotted in Figure 6. The overall mean difference for the whole set of data calculates to 0.12% with a standard deviation (1s) of 0.10%. If categorized according to the year of analysis, the compared data show a mean difference of 0.07 ± 0.10 % for the campaigns in 2007, and of 0.16 ± 0.10% for the campaigns in 2008. For the underlying type of enrichment measurements made with a scintillation detector in the 2nd generation of COMPUCEA, the results demonstrate a remarkable level of performance. Both the observed random and systematic errors are well within the estimated uncertainty of the gamma measurement of about 0.25% for enriched materials, and also well within the uncertainty levels set by the International Target Values.

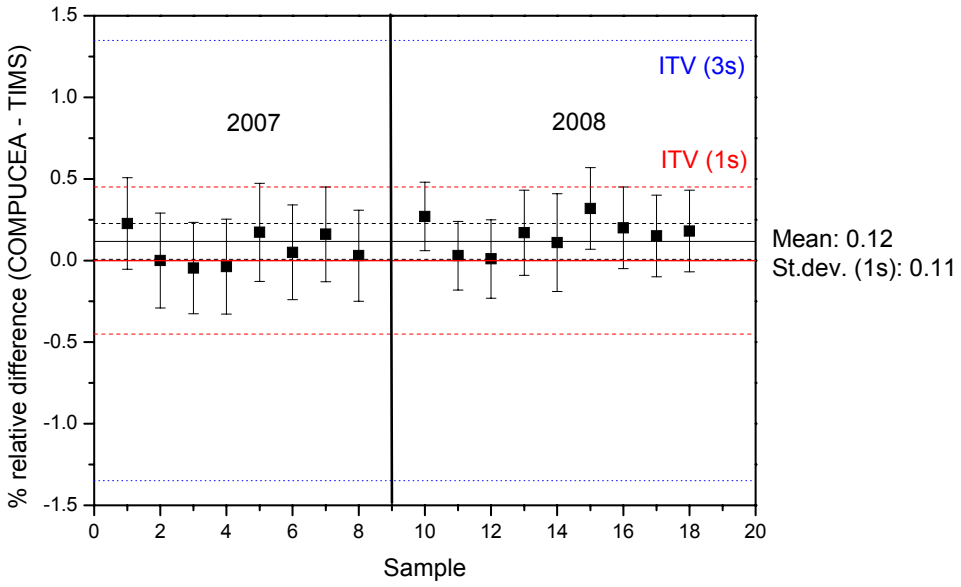

Figure 6: Percentage difference between ^{235}U enrichment values measured with COMPUCEA 2nd generation and TIMS.

4.3. Upgrade for ^{235}U enrichment determination: $LaBr_3$ sandwich detector

In order to further improve the measurement performance of the COMPUCEA equipment, an increase of sample throughput is of relevance during in-field operation, without loss of measurement

performance. For ^{235}U enrichment determination, a counting configuration using two $LaBr_3$ detectors in a sandwich-type arrangement, as shown in Figure 7, promises to significantly enhance the robustness of the counting configuration and the detection efficiency, allowing for shorter counting times and/or improved measurement precision. First experiments were performed using specifically designed sample containers, filled with 10, 12 and 13ml of sample solution. For data evaluation, the two spectra were evaluated separately and the number of counts determined for the 186 keV peaks summed up afterwards. For the 10, 12 and 13 ml samples, an increase of the overall counts of a factor of 2.5, 3 and 3.3, respectively, was obtained, compared to the counting configuration with one detector (and 10 ml sample volume). The reproducibility, both for several repetitions of measurements as well for a series of measurements where the sample container was taken out and put back into position, turned by 180°, was satisfactory, a repetition of 5 samples led to a random uncertainty of < 0.15%.
The new detector configuration (with 13 ml sample volume) will be tested in parallel during the in-field campaigns 2009, the automated evaluation of the 2 gamma spectra needs to be implemented in the COMPUCEA Software package.

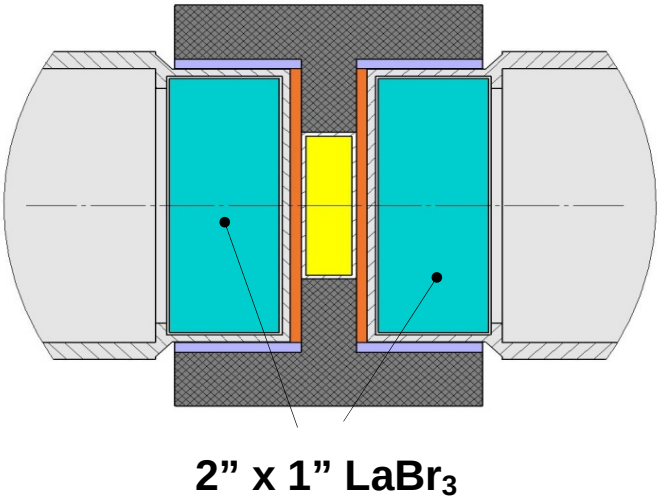

Figure 7: Sandwich detector configuration with two $LaBr_3$ detectors for enhanced efficiency in ^{235}U determination.

5. Conclusion

COMPUCEA is a compact and transportable system which allows high-accuracy uranium elemental assay and enrichment determination from solid uranium samples. The second generation Compucea avoids radioactive sample transport, does not need transport of radioactive sources and attains excellent accuracy with an easily portable system.It is routinely applied in physical inventory verification campaigns at European LEU fuel fabrications plants. The 2nd generation system with a compact L-edge densitometer and $LaBr_3$(Ce) scintillation detection has been evaluated and validated, the performance is well within the International Target Values. A sandwich detector configuration using two $LaBr_3$ detectors for ^{235}U enrichment determination offers improved counting efficiency for higher sample throughput during in-field operation.

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RECENT HKED INSTRUMENTATIONS FOR ANALYTICAL MEASUREMENTS IN CONVENTIONAL AND ADVANCED NUCLEAR FUEL REPROCESSING

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Abstract:

Two Hybrid K-edge Densitometers (HKED's) recently designed at ITU for different applications are presented and described. One of them, designated for installation in the RT-1 reprocessing plant of Mayak, Russia, follows the well-known route of standard HKED applications, i.e. the analysis of process samples from Purex type reprocessing. The second HKED is embedded into a more enlarged non-destructive assay (NDA) station including an additional neutron coincidence counter and a high-resolution gamma spectrometer for the analysis of minor actinides in process samples originating from pyro-reprocessing test facilities at ITU. In addition, the paper also provides an evaluation of new HKED analysis software recently developed at the Los Alamos National Laboratory.

Keywords: Hybrid K-edge densitometer; uranium, plutonium; minor actinides; spectrum evaluation

INTRODUCTION

The Hybrid K-Edge Densitometer (HKED) has become a measurement device of prime importance for analytical measurements in large-scale reprocessing plants, both for Safeguards and for plant process control purposes. Providing a straightforward technical Safeguards tool for timely on-site verification measurements of the uranium and plutonium concentrations in reprocessing input solutions has been one of the main incentives and objectives for its development in the nineteeneighties /1-4/. For this main purpose, but also for other analytical applications, the HKED is meanwhile in routine use in the three major operating reprocessing plants of the world (La Hague - France, Sellafield - U.K. and Rokkasho - Japan).

The Institute for Transuranium Elements (ITU), Karlsruhe, has a long experience in the design and operation of HKED instruments, and in the application of the underlying measurement techniques, viz. K-edge densitometry (KED) and X-ray fluorescence (XRF) analysis. In fact, the majority of the HKED instruments currently in use have been designed and implemented by ITU. Moreover, ITU is routinely operating, on behalf of DG TREN, a number of HKED instruments in the Euratom On-site Laboratories at La Hague and Sellafield, and it is also providing technical support and advices for the operation of ITU-designed HKED instruments in the joint NMCC/IAEA On-site Laboratory at Rokkasho.

This paper describes two special HKED instrumentations recently designed at ITU for two different purposes and applications. One of them, designed for its installation in the RT-1 reprocessing plant at Mayak, Russia, is foreseen for the conventional HKED application, i.e. the analysis of reprocessing input and product solutions. For the second device, designed for in-house use at ITU, the basic HKED setup has been incorporated into a versatile NDA measurement station for the assay of major and minor actinides in a variety of different types of irradiated nuclear materials. Among them, process

samples originating from the ITU pyro-reprocessing test facility will represent a significant fraction of the samples to be analyzed with this multi-purpose system.

Along with the design and operation of different HKED instrumentations, ITU is also constantly involved in the evaluation and adaptation of advanced HKED analysis software. Recent software developed at the Los Alamos National Laboratory, which is likely to become the core of the future analysis software for the HKED, is currently being evaluated in comparison with the analysis codes so far used. Experiences and results from this comparison are also discussed in the paper.

THE HKED FOR RT-1 PLANT, MAYAK PA

In the frame of the project "Modernisation and Enhancement of Nuclear Material Accountancy and Control (NMAC) at the Mayak RT-1 Plant", an undertaking funded by the European Union under his TACIS programme and technically managed by the Joint Research Centre (JRC) of the European Commission in close cooperation with the Russian partners involved, the RT-1 reprocessing plant will be equipped with an HKED for improved uranium and plutonium analytical measurements. The contribution of ITU to this project included so far the design and the survey of the fabrication of the mechanical HKED components, and the execution of performance tests for provisional instrument acceptance prior to the shipment to the Mayak site. This part of the project has been completed in 2008 /5/. Further support will be provided on site during the actual instrument installation and initial calibration, which are both scheduled for 2009.

Instrument design

HKED instruments for the analysis of the highly radioactive reprocessing input solutions need to be attached to well-shielded hot cell facilities designed for the handling of those kinds of materials. In the usual way of implementation the HKED measurement station itself, normally located behind the hot cell, is interfaced via a horizontal shielded sample transfer system to the interior of the respective hot cell. For most of the installations made during the last 10 years the sample transfer system consists of a linear sample changer device /6/. A very important feature, which greatly facilitates the practical HKED implementation, results from the fact that the basic HKED instrument components (X-ray generator, detectors and electronics) remain in an accessible area outside of the hot cell. It is obvious that this feature is of great advantage for easy instrument maintenance.

The standard HKED configuration with a horizontal sample transfer from the hot cell into the instrument had to be modified for the HKED implementation in the RT-1 plant. In order to cope with the specific site conditions, and to cause minimum interference with the existing facilities, the plant operator suggested to locate the HKED assembly on top of the hot cell and not, as normally practiced, directly behind the cell. According to this modification the HKED will be now connected to one of the

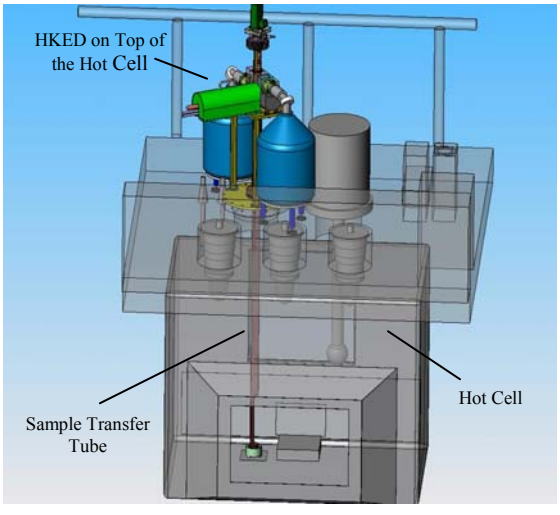

Fig. 1. Overall view of the HKED installation in the RT-1 plant.

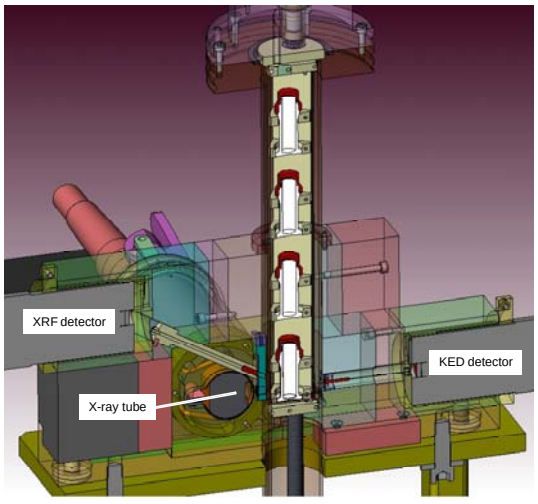

Fig. 2. Vertical cross section of the HKED assembly on top of the hot cell.

shielded hot cells in the high-active laboratory of the RT-1 plant in a way as shown in Fig. 1. The basic HKED assembly is placed on top of the hot cell, while the samples to be analysed are transferred from the working area inside of the hot cell up to the HKED measurement assembly by means of a vertical sample changer.

The sample changer consists of a nearly 3 m long cylindrical stainless steel tube with inner and outer diameters of 56 mm and 61 mm, respectively, which is flanged to an access port in the ceiling of the hot cell. A sample magazine hooked to a spindle inside of the sample changer tube is driven by means of a stepping motor from the conveniently reachable loading point near to the floor in the hot cell up to the measurement position in the HKED assembly. The sample magazine can accommodate up to 4 samples, positioned on top of each other as shown in Fig. 2. The sample containers chosen for the HKED at RT-1 are of the Cogema type. They are identical to those used in the HKED measurements at La Hague and Rokkasho. The cylindrical polyethylene sample containers have an inner diameter of 1.41 cm with a maximum capacity of about 8 ml. For the HKED measurements the sample containers should be filled with a minimum of about 4 ml of sample solution.

The HKED system for RT-1 is assembled with the standard HKED components, i.e. a 160 kV/3 kW X-ray generator powering a 160 kV metal ceramic X-ray tube, two planar HpGe detectors (200 mm^2 x 10 mm), an Alpha Workstation with the dedicated HKED software controlling the data acquisition and sample changer operation, and performing the spectrum analysis and data evaluation. The standard components were provided by Canberra.

The arrangement of the X-ray tube, the KED and XRF detectors in the tungsten shielding and collimation assembly is shown in the cut-out view of Fig. 2. The shortest possible distance of 65 mm between the focal spot of the X-ray tube and the centre of the cylindrical sample, realized in all previous HKED systems, has been also maintained in the present set-up. This also holds for the standard viewing angle for the XRF detector of 150° relative to the primary X-ray beam directed from the X-ray tube towards the sample. With the use of a standard detector cryostat (horizontal dipstick) this particular configuration implies that the fluoresced X-rays hit the XRF detector surface at an angle of 30°. A practical benefit of the vertical sample changer configuration, where the samples are moved in the plane subtended by the horizontal primary X-ray beam axis and the vertical cylinder axis of the sample containers, results from the fact that the vertical positioning of the cylindrical sample containers relative to the X-ray beam axis is not very critical.

HKED applications at RT-1

The main applications of the HKED at RT-1 plant will be similar to the ones in other reprocessing facilities, i.e. the major use will be dedicated to the analysis of the input and product solutions. Fig. 3 shows a schematic flow diagram for the RT-1 plant with a specification of the expected typical uranium and plutonium concentration levels in the relevant process solutions. The major type of spent fuel reprocessed at the RT-1 plant originates from VVER-440 reactors (with initial uranium enrichment ranging from 3-4.4%), which makes up about 90% of the plant throughput. A smaller fraction of reprocessed spent fuel comes from BN-600 reactors (with initial enrichments in the range of 15-26%).

The dissolver solutions of the reprocessed VVER-fuels are fairly high in concentration (somewhat higher than normally encountered in the HKED measurements on reprocessed LWR fuels). Good performance for the U-concentration determination from KED, and for the U/Pu ratio measurement from XRF, can be assured for this type of solution. The graph in Fig. 4 shows that for uranium concentrations above 200 g/l the KED measurement precision[1] reaches a level of 0.2% for the given type of sample container with a path length of 1.41 cm. The given precision values refer to a counting time of 1000 s. Within the same counting time the U/Pu ratio measurement by XRF provides a precision of about 0.8% for an assumed U/Pu ratio of about 100 (see Fig. 5). This precision value for

[1] All precision values given are relative 1s

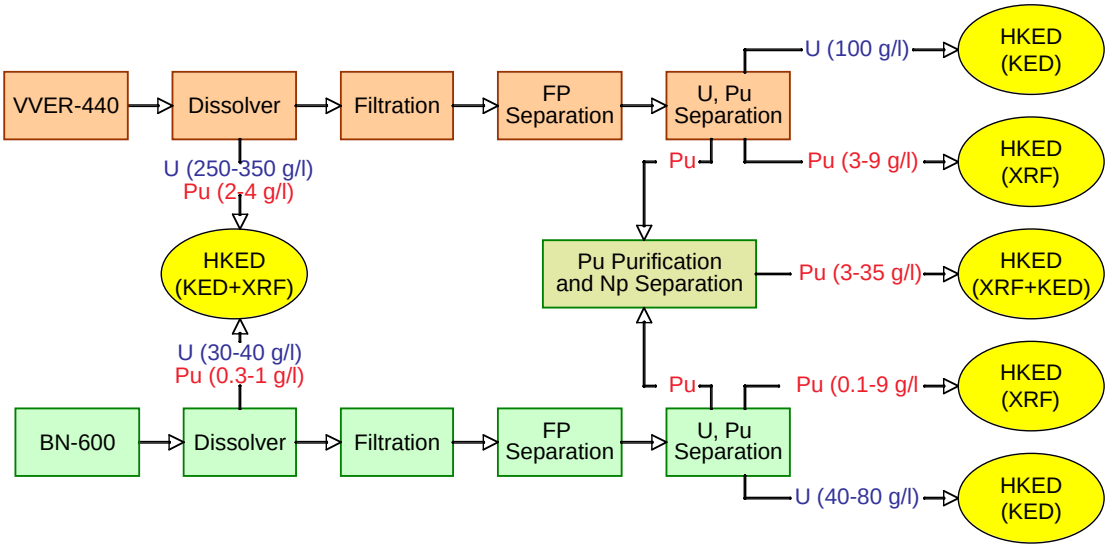

Fig. 3. Anticipated HKED applications at RT-1.

the U/Pu ratio can be of course improved with a longer counting time. The adopted standard practice in most of the HKED applications is to perform at least 3 repeat measurements of 1000 s counting time each for improved measurement statistics.

The dissolver solutions from reprocessed BN-600 fuels are significantly lower in concentration (see Fig. 3). Nonetheless, the standard HKED measurement approach, i.e. determining the U-concentration from KED, and the Pu-concentration from the combined KED+XRF measurements, can also be applied to this type of solution. The measurement performances, however, will be somewhat inferior to the ones obtained for the VVER case. Under the same measurement conditions as stated before the KED precision for uranium will rise to a level of about 1% at the expected uranium concentration level of about 30-40 g/l, and the precision for the U/Pu ratio to about 1.5-2%. For this type of samples, counting times longer than 1000 s are definitely recommended to improve measurement precision.

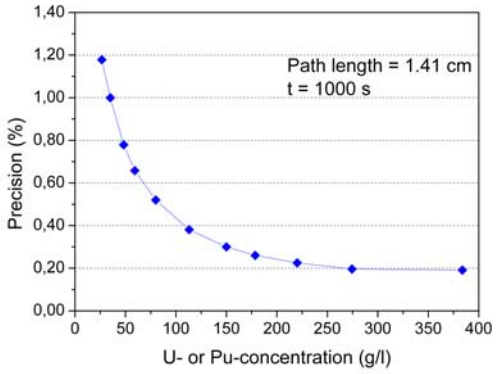

Fig. 4. Typical KED measurement precision for U and Pu versus their concentration.

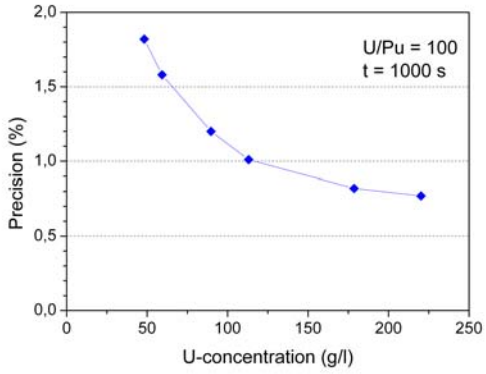

Fig. 5. Typical measurement precision for the U/Pu ratio versus the U-concentration (U/Pu ratio ≈ 100).

The proposed type of measurement for the analysis of product samples from the output of the plant will depend on the given concentration level. For the determination of the uranium concentration, both for the VVER and the BN fuel offering concentrations in the range from about 40-100 gU/l, K-edge densitometry will be the preferred mode of analysis because of the high reliability of the method. A switch from KED to XRF is normally recommended for concentration levels from about 20-30 g/l

downwards. This will apply for the majority of the plutonium product solutions from the RT-1 plant according to the stated concentration levels in Fig. 3.

For the intermediate concentration levels in the range from 20-40 g/l the XRF analysis provides the by far higher measurement precision compared to the KED measurement. On the other hand one has to be aware of the fact that the analysis made by XRF is much more prone to potential systematic measurement errors than the KED technique. For improved measurement confidence, the complementary use of both techniques is therefore a good measurement practice for samples in this intermediate concentration range. For the analysis of samples with concentrations below 20 g/l, however, the XRF branch of the HKED remains the only option. The useful range of applicability providing still acceptable measurement precision extends down to concentration levels around 0.2 g/l, both for U and Pu. The measurement precision achieved with a counting time of 1000 s for concentrations around 1 g/l is typically about 1%.

HKED-NCC-HRGS MEASUREMENT STATION AT ITU

ITU is pursuing a substantial research programme devoted to the partitioning and transmutation (P&T) of spent fuels. This includes both investigations on advanced processes for improved actinide and fission product separation, and the development of special fuels for transmutation /7/. Among the different approaches for partitioning, major efforts are being directed towards the development and testing of pyrochemical reprocessing techniques. For this purpose dedicated test facilities for studying pyro-reprocessing with non-irradiated test fuels as well as with real fuels have been set up at ITU.

The ongoing research work on the pyro-reprocessing technology requires continuous analytical support for (timely) analysis of the various process samples in order to verify and to assess the quality and efficiency of the respective separation processes. Over the years some of this analytical support, especially for minor actinide analysis, has been also provided by various radiometric techniques as far as they appeared applicable. The radiometric techniques used for this purpose at ITU included mainly X-ray fluorescence analysis (XRF), high-resolution gamma spectrometry (HRGS), passive neutron coincidence counting (PNCC), and occasionally also calorimetry /8, 9/. The measurements were made with instrumentations coupled to gloveboxes, which allow the handling of non-irradiated nuclear materials. The analytical support with the radiometric techniques was therefore possible as long as the investigations for pyro-reprocessing were conducted with these type of materials.

Meanwhile the research work at ITU on the pyro-reprocessing techniques is proceeding to experimental studies, which also include actual irradiated nuclear fuels. In order to be able to provide further analytical support for this new type of highly radioactive process specimens, the measurement environment for the radiometric analysis techniques needed to be adapted accordingly. This has led to the conception and practical realization of a versatile NDA measurement station for the analysis of irradiated fuel samples.

Layout

The NDA station has been designed for installation at the hot cell facilities of ITU. Fig. 6 shows the general layout of the installation. The NDA station, displayed in Fig. 7 in closer detail, is located at the rear of one of the ITU hot cells. The sample transfer from the interior of the hot cell into the shielded measurement station is accomplished by means of a specially designed sample transfer system. It physically consists of a double-bent stainless steel tube, in which the samples to be measured are pulled up from the hot cell to two different measurement positions in the NDA station. The sample lift-up is effected electrically with a servo motor, which is winding up a thin stainless steel cable with the sample container fixed to its bottom end. Since the sample transfer tube represents an open connection to the interior of the hot cell, a small glovebox for protection against contamination is sitting on top of the NDA measurement station (see Fig. 6). It is coupled to the sample transfer tube via a sealed flange. The glovebox houses the driving assembly, and it allows access for its maintenance, if needed. The photograph in Fig. 8 shows the complete mechanical assembly as it will be bolted to the biological shield at the rear of the hot cell. After the installation, which will occur after

the completion of a cleaning up of the respective facility, the steel frame containing the NDA measurement devices will be shielded with a 5-cm thick lead shield for radiation protection.

Sample manipulations required prior to the measurements inside of the hot cell are carried out by means of telemanipulators. The respective operations like the aliquoting of sample liquids into special sample vials and the loading of the samples into the sample carrier of the sample transfer system can be observed through leaded glass windows from the front side of the hot cell. On their drive out from the hot cell into the measurement station the samples will first reach a neutron coincidence counter, which is surrounding a 40 cm long section of the sample transfer tube just behind the biological shield of the hot cell (see Fig. 7). An electromagnetic sensor defines the sample stop in the centre of the counter. The polyethylene body of the neutron counter is equipped with a total of 16 ^{3}He tubes (25 mm outer diameter x 300 mm active length with 6 bar ^{3}He and 1 bar Ar). Each of the ^{3}He tubes is provided with its own preamplifier and their logical outputs are "OR-ed" before transmission to the Total and Coincidence neutron detection electronics. Monte Carlo simulations were carried out to define for the given number of ^{3}He tubes the optimum counter configuration ensuring the highest detection efficiency. This has been experimentally measured to be 31% for ^{240}Pu fission neutrons. The sample transfer tube passing through the centre axis of the neutron counter is surrounded by a 2.8 cm thick cylindrical tungsten sleeve to provide protection for the ^{3}He tubes against the intense gamma radiations from fission products and the tubes themselves are constructed to operate in gamma radiation fields of up to 100 R/h.

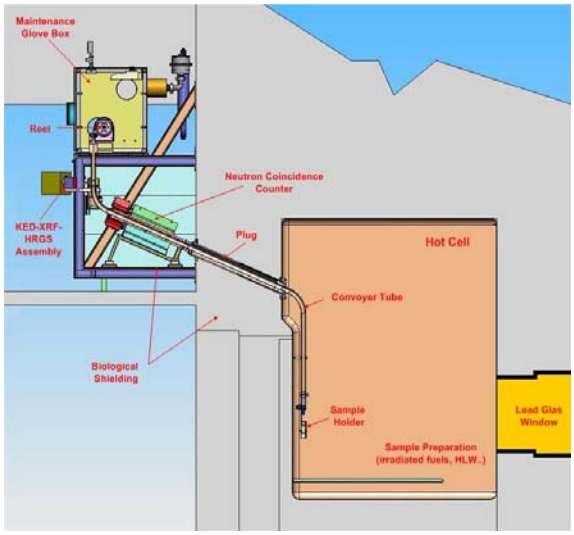

Fig. 6. General layout of the installation.

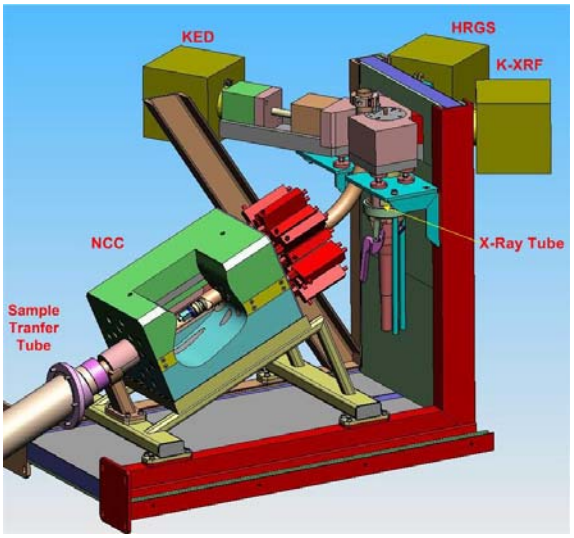

Fig. 7. Measurement components of the NDA station.

A second sample stop actuated by another sensor in the top end section of the transfer system positions the samples in a combined KED-XRF-HRGS measurement assembly (see Figs 7 and 9). As shown in the cross-sectional view in Fig. 9 this assembly contains 3 HpGe detectors (electrically-cooled) and a Peltier-cooled high-resolution (FWHM = 140 eV at 5.9 keV) Si detector together with a 160 kV metal ceramic X-ray tube with tungsten target. The K-edge and XRF1 detectors, both of planar type (200 mm^2 x 10 mm) are forming together with the X-ray tube the standard HKED configuration. The 3rd (coaxial) HpGe detector (58 mm diameter x 32 mm) serves for high-resolution passive gamma spectrometry applications.

The Si drift detector has been added for optional high-resolution XRF spectrometry at lower X-ray energies down to about 10 keV. To allow those X-rays to escape from the samples, the stainless steel material of the sample transfer tube was replaced by a short tube section made from low-Z material (PEEK) with a relatively thin transmission window.

Fig. 8. Photograph of the compact NDA station (Ge detectors removed) prior to the installation at the hot cell.

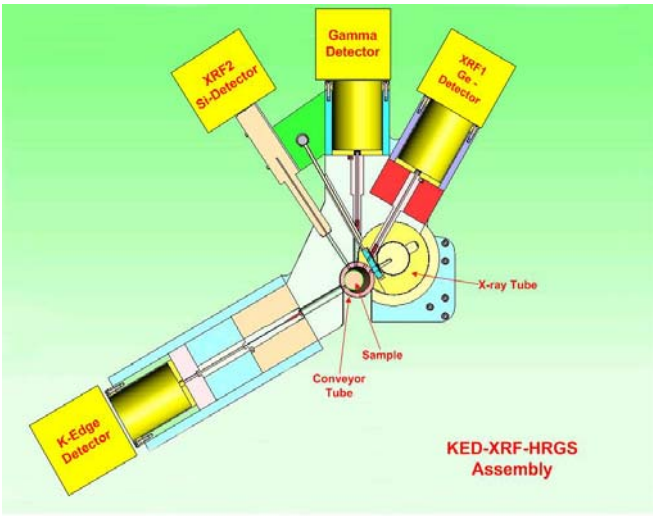

Fig. 9. Detector configuration in the KED-XRF-HRGS measurement station.

Applications

Apart from its envisaged main application of analyzing process samples from the pyro-reprocessing studies on irradiated nuclear fuels, the NDA measurement station will provide also measurement options for the analysis and accountancy of all kinds of nuclear materials handled for research work in the ITU hot cell facilities. It is expected that the XRF measurement capabilities, together with gamma spectrometry, will take over the major part of the analyses.

Samples taken from the molten salt (LiCl-KCl) and metal phases (mostly Bi or Cd) of the pyro-reprocessing process readily dissolve in nitric acid, which helps for the analysis. However, the sample quantities typically taken are relatively small (of the order of 100 mg), which limits the available amounts for analysis by the radiometric techniques. For minor actinides the K-XRF part of the HKED can provide analysis capabilities for any sample type in liquid form with an actinide mass fraction larger than about 0.02 %, or, in absolute quantities, of about 50-100 µg for the individual actinide element /9/. An example of a typical K-XRF spectrum is displayed in Fig. 10. Because of the strong irradiation power of the X-ray tube this method can tolerate fairly high levels of gamma activities from fission products (up to 10^{10} Bq/g) in the samples under assay.

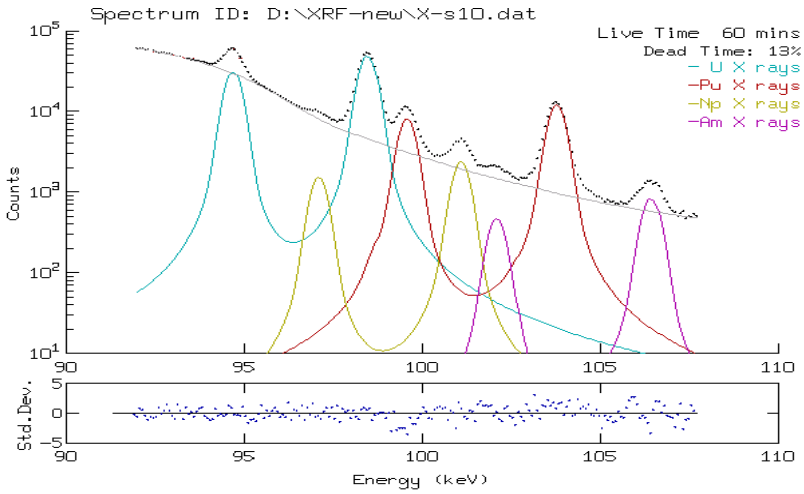

Fig. 10. XRF spectrum from a dissolved molten salt sample (non-irradiated fuel).

The K-XRF measurement of the HKED also opens a measurement window for the analysis of lanthanide elements, which are of interest in the fuel partitioning as well. The K X-rays of the lanthanides with energies in the 30-45 keV range (Fig. 11) are falling into a spectrum region of relatively low Compton background in the XRF spectra produced under the measurement conditions of the HKED. This leads to reasonably low detection limits of about 10 ppm for those elements.

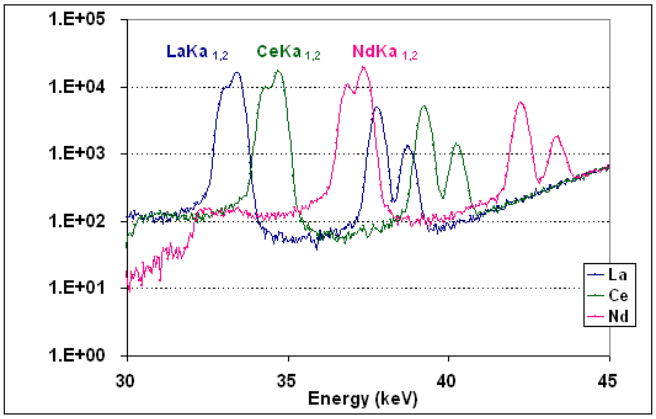

Fig. 11. K-XRF response from lanthanide elements.

For the quantitative assay by XRF the effect of photon attenuation, both for the interrogating primary X-ray beam from the X-ray tube and for the fluoresced characteristic X rays used for analysis, has to be carefully considered. This holds in particular for the analysis of the actinide elements embedded in the molten metal samples with high-Z elements like Bi as matrix element, and for the lower X-ray energies of the lanthanide elements. In order to limit and to reduce the necessary calibration efforts, the measurements are supported by MCNP-based simulations to reliably calculate the XRF response in dependence of varying sample composition and measurement conditions /10/.

The application of the neutron coincidence counter in the NDA station is mainly dedicated to the determination of Cm. Neutron coincidence counting (NCC) offers the best choice for the direct quantitative assay of this element in practically all kinds of process samples of interest, solid or liquid. The measurement technique is truly non-destructive, and matrix effects play a minor role for quantitative assay. NCC can measure amounts of Cm down to about 100 ng. Assuming that ^{242}Cm ($t_{1/2}$ = 163 days) has normally decayed away, ^{244}Cm will then provide the main contribution to the measured neutron response, i.e. about 95% for Cm from LWR fuel and about 90% for Cm from FBR fuel. ^{246}Cm will contribute the remaining 5-10%. This situation also holds for samples containing

plutonium, where Cm still remains the main neutron emitter (≥ 95%) for Pu/Cm ratios up to about 1000.

High-resolution gamma spectrometry can be used for the determination of certain actinides, where XRF is not sensitive enough, and of course also for the identification and quantification of all gamma-emitting fission products. Among the minor actinides easiest measured by HRGS is Am (^{241}Am and ^{243}Am), which can be detected even in the presence of a significant gamma background from fission products. By contrast, the determination of the minor actinide element Np (^{237}Np) would require a prior separation of the gross fission product activity before the gamma measurement /11/.

EVALUATION OF HKED SOFTWARE

A proper mechanical instrument design, high-quality and high-resolution Ge detectors and advanced and optimally adjusted pulse processors are among the basic hardware requirements needed to produce high-quality spectral data in the K-edge and K-XRF measurements of the HKED. Equally important, however, is the application of the appropriate software to correctly extract the desired spectral information from the acquired spectra. For the K-edge measurement this basically means the correct determination of the photon transmission ratio across the K-absorption edge of the analyte, and for the K-XRF measurements the correct determination of the net peak areas of the fluoresced X-rays of interest. In both measurements the software also has to take properly into account the impact of known physical processes on the measured spectral responses, which depend on the actual sample composition.

The HKED analysis software originally developed at the Kernforschungszentrum Karlsruhe had been designed for some well-defined basic applications like the analysis of measurements made on the input and product solutions from a reprocessing plant. The software had been licensed to Canberra, which embedded the developed basic algorithms for spectrum analysis into his Genie-VMS architecture to create a complete HKED software for measurement control, analysis, calibration etc. The respective HKED software is running under the DEC-OpenVMS operating system. Over the years it has been continuously refined and extended to cope also with additional applications. To this end a variety of different ‘measurement protocols’ have been created for the user. Among the recognized limitations of the existing software one should mention at least 3 features: (i) the user must have knowledge about the nature and approximate composition of the sample to be analysed, (ii) the peak area analysis for the XRF spectra employs the traditional region-of-interest (ROI) method, which severely limits the versatility and capability for multi-element analysis, and (iii) also the practical aspect that the software is running under the not so common OpenVMS operating system.

More than 10 years ago the Los Alamos National Laboratory (LANL) started development work for a new Windows-based HKED software. Part of this work has been carried out under the US support programme to the IAEA. The basic idea and objective behind this development was to create an analysis software allowing automated spectrum analysis without prior knowledge of the nature and composition of the sample. New analysis tools introduced for this purpose included a generalized multi-edge fitting technique for the analysis of the K-edge spectra, and the use of response function fitting for the analysis of the XRF spectra /12/. The new software will shortly become available commercially.

In collaboration between LANL, IAEA and ITU the new LANL HKED software has already undergone some preliminary testing and performance evaluation at the various stages of its development. A somewhat more extended performance evaluation, in comparison with the previous Canberra code, has been recently made at ITU. This comparison addressed one of the key applications of the HKED, i.e. the analysis of reprocessing input solutions. Available KED and XRF spectra from measurements performed on a larger number (about 80) of typical input solutions from LWR fuels have been comparatively analysed with the Canberra and LANL analysis codes. The obtained results for the uranium concentration and for the U/Pu ratio were then compared with the results from Isotope Dilution Mass Spectrometry (IDMS) analyses.

The calibration parameters required by the two codes were evaluated from a set of calibration spectra acquired from 6 reference solutions. The reference solutions were prepared from two different input solutions with U/Pu ratios of 80 and 116, respectively. From each input solution two further dilutions were made to vary the absolute U and Pu concentrations at a fixed U/Pu ratio (see Table 1). The reference values for the U and Pu mass fraction in the two mother solutions were determined by IDMS. The conversion from mass fraction to element concentration then follows from an accurate determination of the solution density. The calibration measurements with this set of reference solutions were made a few weeks before the following series of input solutions measurements, which extended over a period of about 6 months.

Results for uranium from KED

Table 1 summarizes the results of the KED calibration for uranium. The data are plotted in Fig. 12 for better illustration. Both codes provide a similar pattern of results for the given set of calibration samples. The scatter of the data points as expressed by the values for the relative standard deviations given in Table 1 are of comparable magnitude (0.12 *vs* 0.18%, respectively). Since the obtained calibration data did not provide conclusive evidence for the existence of a potential non-linearity of the instrument response in dependence of the uranium concentration, and since the individual calibration points did not deviate by more than ± 0.25% from the mean value, a constant K-edge calibration factor (Δμ) has been adopted for both codes for the given range of uranium concentrations.

Table 1. Calibration results for U from KED.

					Canberra code		LANL code	
Sample	No. of repeats	U(Decl.) (g/l)	U/Pu Decl.	1s-Uncert. For U (%)	Meas/Decl (Mean)	RSD (%)	Meas/Decl (Mean)	RSD (%)
1	3	199.36	80.004	0.06	0.9992	0.16	0.9991	0.21
2	3	144.63	80.004	0.06	1.0009	0.13	1.0027	0.17
3	3	78.68	80.004	0.06	0.9998	0.31	0.9989	0.57
4	3	219.29	116.34	0.06	0.9999	0.08	0.9987	0.14
5	3	152.68	116.34	0.06	1.0018	0.36	1.0021	0.21
6	3	81.25	116.34	0.06	0.9983	0.24	0.9986	0.04
				Mean	1.0000	0.21	1.0000	0.22
				RSD (%)	0.12		0.18	

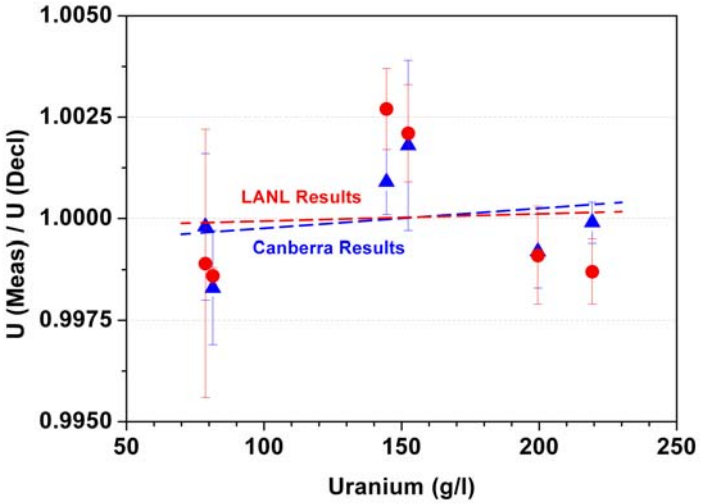

Fig. 12: Plot of Measured-to-Declared U concentrations from the KED calibration.

With the Canberra and LANL codes calibrated in this manner, K-edge spectra from measurements on 80 input solutions were then comparatively analyzed with both analysis codes. The majority of the analyzed input solutions had uranium concentration between about 180 and 200 g/l. The results are plotted in Fig. 13 as ratios 'Measured-to-Declared'. The data show an unexpected average bias of about 0.5% for the LANL code, while the results obtained with the Canberra code, on the average, turn out to be nearly unbiased. The reason for the bias in the LANL data is still being investigated. Apart from this bias, the set of results obtained with the two analysis codes show a comparable scatter with relative standard deviations of 0.46% (Canberra) and 0.49% (LANL).

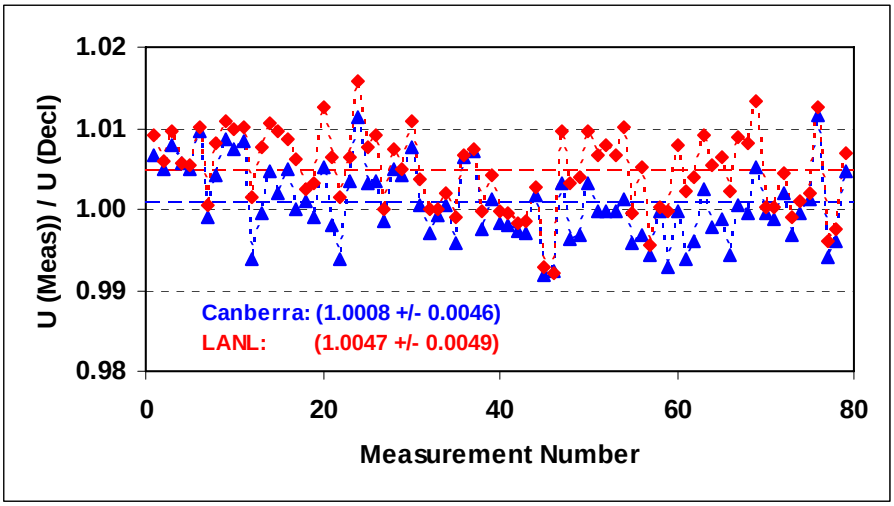

Fig. 13. Plot of Measured-to-Declared values for the uranium concentration evaluated with the Canberra and LANL HKED analysis codes from input solution measurements.

Results for the U/Pu ratio from XRF

The determination of the U/Pu ratio from the XRF measurement made with the HKED is not as simple and straightforward as the KED measurement of the uranium concentration. In fact, energy-dispersive XRF measurements made at element ratios of the order of 100 represent a challenging task. At this large ratio, the peak-to-background ratio of the main X-ray peak ($K\alpha_1$) of the minor element plutonium normally takes values of less than 1 under the HKED measurement conditions. This situation makes it very difficult to evaluate truly unbiased net peak areas for the plutonium X-ray peak, and hence for the derived U/Pu ratio.

Despite a lot of efforts invested into the development of the analysis code it has been so far not possible to make the U/Pu ratio determination from the HKED XRF spectra completely immune to variations in certain sample parameters like the uranium and plutonium concentration levels or the magnitude of the U/Pu element ratio itself. This becomes evident from the plots of the calibration data in Figs. 14 and 15, where the measured-to-declared ratios from the U/Pu-ratio calibration measurements are plotted once versus the uranium concentration (Fig. 14), and once versus the U/Pu ratio (Fig. 15). The present software for the U/Pu calibration allows for non-linearity corrections which are not applied to the data displayed in Fig. 14 and 15.

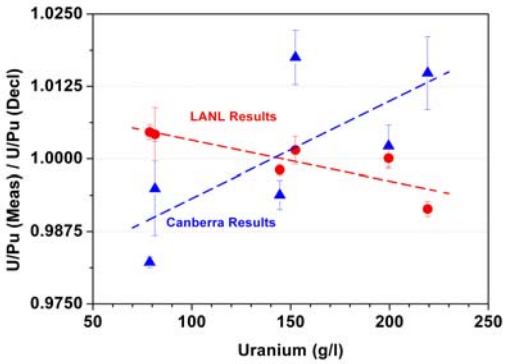

Fig. 14. Calibration results for the U/Pu ratio versus the uranium concentration.

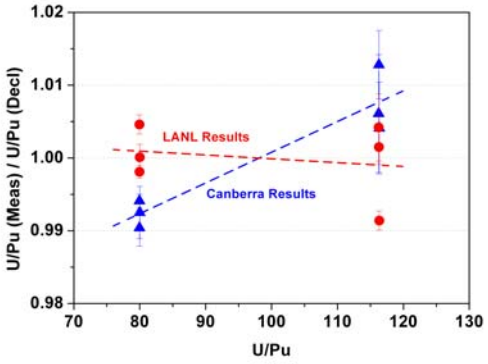

Fig. 15. Calibration results for the U/Pu ratio versus the U/Pu element ratio.

From a comparison of the calibration data obtained with the two analysis codes we make the following observations:

(i) On the average, the measurement precision of the LANL results is about a factor of 2 better than the precision of the Canberra results (0.35% compared to 0.75%, see Table 2).

(ii) Both codes manifest a dependence of the measured U/Pu ratio on the uranium concentration, but of opposite direction. However, the LANL results turn out to be less sensitive (about a factor of 2) to this parameter than the results obtained with the Canberra code.

(iii) The results of the Canberra code also show a (significant) dependence on the U/Pu ratio, while the LANL results appear to be insensitive to this parameter at least for the range of U/Pu ratios (80 and 116) covered by the calibration samples.

Table 2. Calibration results for the U/Pu ratio from XRF.

					Canberra code		LANL code	
Sample	No. of repeats	U(Decl.) (g/l)	U/Pu Decl.	1s-Uncert. for U/Pu (%)	Meas/Decl (Mean)	RSD (%)	Meas/Decl (Mean)	RSD (%)
1	3	199.36	80.004	0.07	0.9925	0.62	1.0001	0.31
2	3	144.63	80.004	0.07	0.9904	0.44	0.9981	0.15
3	3	78.68	80.004	0.07	0.9941	0.15	1.0046	0.22
4	3	219.29	116.34	0.08	1.0041	1.09	0.9914	0.22
5	3	152.68	116.34	0.08	1.0128	0.82	1.0015	0.42
6	3	81.25	116.34	0.08	1.0061	1.40	1.0042	0.80
				Mean	1.0000	0.75	1.0000	0.35
				RSD (%)	0.89		0.49	

The results for the U/Pu ratio values evaluated from the set of the 80 input measurements, displayed again as ratios Measured-to-Declared, are plotted in Fig. 16. The first 40 samples of the measurement series had U/Pu ratios in the range between 90 and 100, while the last 40 samples had somewhat higher U/Pu ratios in the range between 100 and 120. A differentiation was made between the two sub-series for the Canberra results because of their known dependence on the U/Pu ratio (Fig. 15), whereas the data from the LANL code not showing this dependence were treated as one single set. The obtained mean values and their standard deviations for the respective measurement series are quoted in the figure. For the full set of data a very small average bias of 0.12% is observed for the LANL results, whereas the results from the analysis with the Canberra code, even after their differentiation according

to the U/Pu ratio, show significant average biases 0.59% and 1.35%, respectively, for the two ratio categories. The observed standard deviations turned out to be about the same (0.7%) for all evaluations.

In general the results from the present performance evaluation suggest that the method of response function fitting applied in the LANL code is indeed superior to the traditional ROI method for peak area evaluation applied in the Canberra code. As mentioned before, the method of response function fitting also provides much more flexibility and versatility with respect to multi-element analysis for samples deviating in composition from those normally obtained in the PUREX reprocessing of LWR fuels (see example shown in Fig. 10). Another benefit offered by the method of response function fitting is the high degree of insensitivity of this kind of analysis to the additional variable radiation background produced by fission products. In order to eliminate the impact of the FP radiation in the XRF analysis of input solutions with the ROI method it became a standard practice to acquire from each input solution also a corresponding passive gamma spectrum and to subtract this from the XRF spectrum. This additional measurement and analysis step can be abandoned with the application of the response function fitting technique. This finding has been already substantiated in previous tests and performance evaluations made with the LANL software.

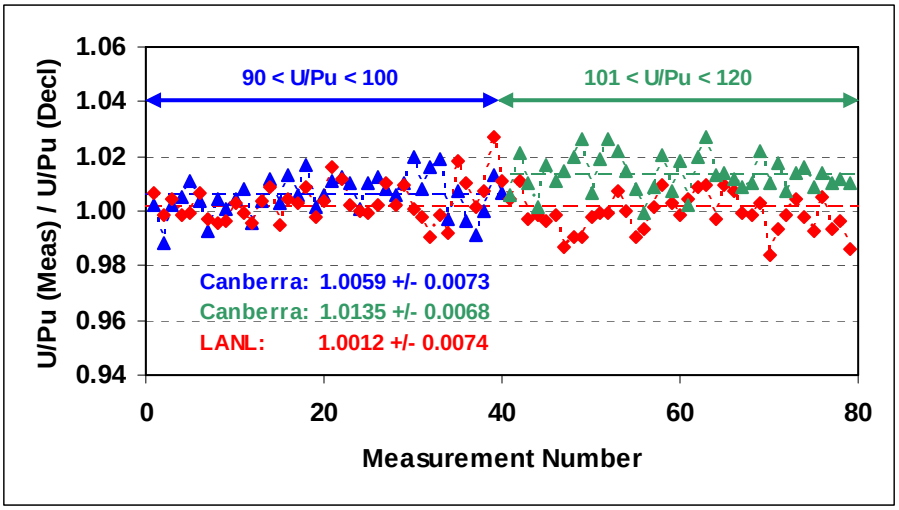

Fig. 16. Plot of 'Measured-to-Declared' values for the U/Pu ratio evaluated with the Canberra and LANL HKED analysis codes. Quoted figures are mean values and their standard deviations.

OUTLOOK

Examples given in this paper illustrate that the HKED measurement technology can be advantageously extended to applications beyond the scope of its originally intended use. For this extension the design of appropriate instrumentations in terms of hardware, adapted to the respective requirements and measurement situations, normally represents the least challenge. More emphasis and efforts, however, will have to be put in the near future on the development of appropriate software tools for spectrum analysis. The increased and efficient use of methods for the simulation of the different measurement situations also represents another area deserving more and more attention.

Acknowledgement

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A New Robotic Arm System for Separation of High-Active Liquid Waste and Spent Fuel Samples at the IAEA Safeguards Analytical Laboratory, Seibersdorf, Austria

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Abstract:

A broad range of radioanalytical measurements are carried out in support of nonproliferation treaty verification, commercial nuclear applications, nuclear waste processing, environmental monitoring, cleanup and restoration of contaminated sites, and national and global security applications. Where measurement precision and accuracy must be of very high quality, wet radiochemical processing techniques such as chemical separation are still required before measurement of the sample. Often, the sample preparation and chemical separation steps require tedious and labor intensive attention by the chemist. Automating these steps decreases the dose to the chemist, and increases the quality of the work by minimizing human-caused errors. Here we describe a new robotic arm system recently installed at the IAEA Safeguards Analytical Laboratory (SAL) for performing chemical separations of high-active liquid waste solutions and spent fuel samples. Based on previous experience with robotic systems at SAL, it is expected that the new robotic arm system will result in decreased failure rates on sample analysis, increased measurement repeatability, and higher sample throughput compared to manual operations. The robot automation system in SAL utilizes a CRS-Catalyst-5 arm and several electronic modules for controlling pumps, hotplates, IR-switches and other devices. A custom-made separation unit accommodates 8 columns, 16 fraction vials and 8 waste collectors. Two separation units will be used for simultaneous separation of 16 samples. A barcode reader is incorporated into the system for sample tracking. State-of-the-art software techniques such as .NET 2.0 development and MS SQL Server database (utilizing stored procedures) were applied to reliably control the robotic arm (COM component) and the associated electronic devices (ADAM .NET library). The custom-made software application emphasizes reliability as well as automatic continuation after unexpected shut downs. Furthermore the software application provides flexibility regarding changes in the chemical procedures.

Keywords: robotic handling, spent fuel, radiochemistry

1. Introduction

The IAEA Safeguards Analytical Laboratory (SAL) is the primary laboratory in the Agency's network of analytical laboratories (NWAL) used for the analysis of nuclear material samples. Wet radiochemical processing techniques, such as chemical separation of U, Pu and fission products are routinely applied in SAL to all samples of spent fuel and high-active liquid waste. The number of such samples fluctuates year to year from dozens to hundreds, according to facility operations and the concomitant inspection goals and schedules. The samples are received in solid form and go through a series of

steps to dissolve and separate unwanted compounds. Manual processing of these radioactive samples is laborious and time consuming, exposes laboratory analysts to unnecessarily high dose rates, and increases the chance for errors during the setup and handling of the samples and associated quality control materials. Robotic handling systems significantly reduce the amount of manual processing, and thus move the lab closer to the ALARA principle, while also improving the overall quality of the sample analysis process by reducing the number of human-errors.

There is a long track record in SAL of using automated systems in the field of spent fuel sample analysis. First prototypes were developed in the early 1990s and soon thereafter put into operation[1]. Later this system design was upgraded with new hardware components, a new robot and a complete rewrite of the software allowing it to run in a Microsoft Windows environment. A further improvement was undertaken to develop the prototype of an even more flexible automation system for a robotic setup at the On-Site Laboratory at Rokkasho, Japan[2].

The latest development described here is a new generation robot arm system for SAL that takes advantage of prior operational experience and modern robotics and computer control systems. The combination of these advances will greatly ease the setup and run processes of the system, and will provide a simplified platform on which to make continuous software modifications and/or improvements in the future. In addition, a customized glove box developed in SAL encloses the system, and takes full advantage of the large working area of the robot arm. The complete system will be capable of routinely processing the chemical separation of 16 samples per working day, or up to 32 spent fuel samples per 24 hours. Furthermore, the system will be capable of non-supervised operation in compliance with the safety codes of the laboratory. These improvements will increase the sample throughput by 50-100% and will allow staff to work on other tasks (e.g., method improvement and development) that normally might not be possible because of time constraints.

2. Description of the Robotic System and Controllers

The core of the automated system is a five-axis robot arm manufactured by Thermo/CRS called the CataLyst-5 (Figure 1). The robot arm will perform all tasks which are normally performed by a human operator. It will move vials, pick up dispensers and handle pipette tips for the transfer of samples and small quantities of reagents. Additional tasks will include the movement of a rack into the appropriate position for fraction collection at the separation unit, and the closing and opening of a hood above the hot plates.

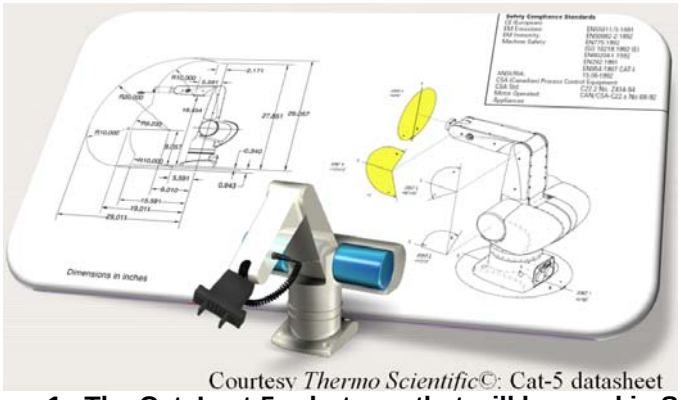

Figure 1. The CataLyst-5 robot arm that will be used in SAL.

A number of supporting components were designed and manufactured in house for autonomous operation of the chemical separation procedure (Figure 2). Piston pumps will be used for delivering precise quantities of the required reagents via a multitube dispensing system. One additional piston pump will be used for handling pipetting actions, such that the robot will manipulate a custom-made pipette holder to pick up pipette tips and draw and dispense solutions. An electric mixer (shaker) was installed as well as hotplates with supporting infrared lamps. A peristaltic pump is used to deliver waste solutions to a waste container. To avoid corrosion, a barcode reader was installed outside of the glove box supported by the peristaltic pump, which is used as a rotating device for barcode reading on vials. The glove box wall does not affect the efficiency of the barcode scanning laser.

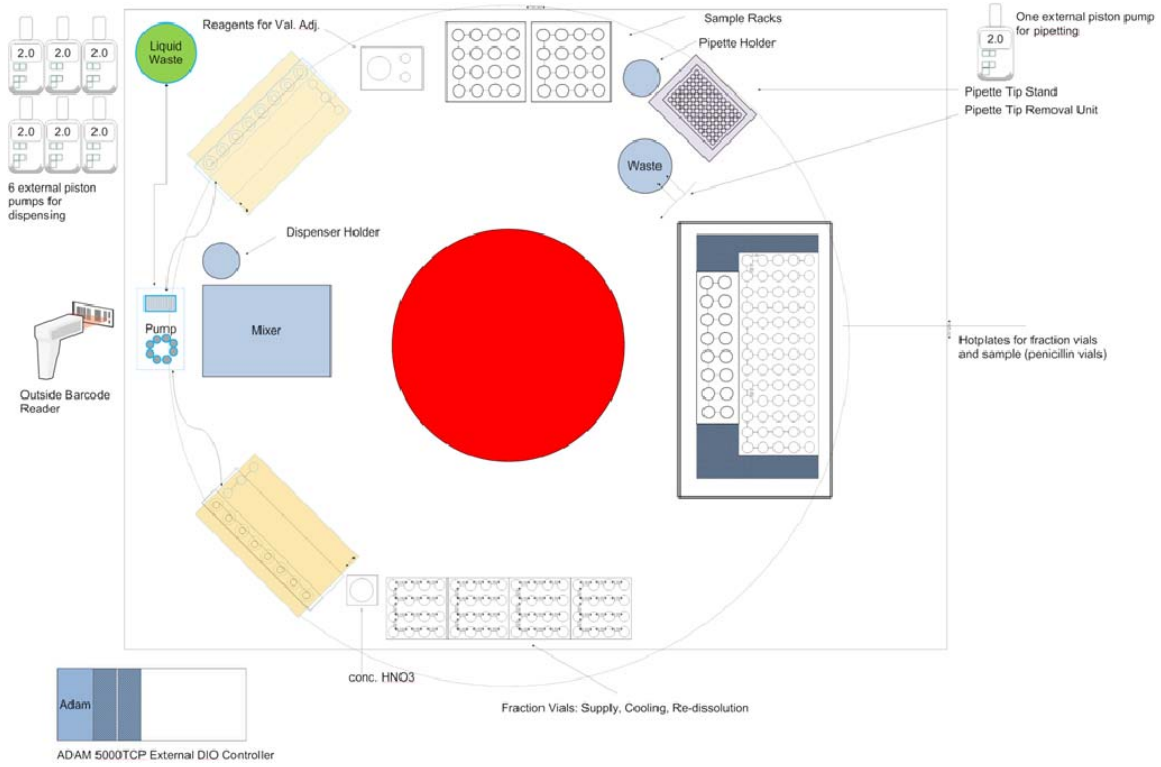

Figure 2. Glove box layout showing the position of the robot arm (circular area) and associated components.

The communication with all devices, whether they are located outside of the glove box (pumps, barcode reader) or inside the glove box, are handled via serial communication (COM port) or via an electronic controller allowing digital and analog inputs and outputs (ADAM™ from Advantec). A set of 16 digital inputs is used to verify the separation process at the chromatographic columns by counting the drops and estimating the volume which was passing through.

Cooling racks for vials and heating racks made out of Aluminium and covered with Teflon (to control corrosion) were designed and fabricated in house. The heating racks and hot plates are enclosed in a stainless steel casing that is connected to a separate venting system in order to mitigate the build up of acidic vapours in the glove box. A removable hood was included so the robot can transport vials to and from the hot plates. Several IR lamps inside the hood will support the sample evaporation process

Finally at the core of the separation procedure is the actual separation task performed on TOPO/Silicagel columns[3,4]. In this respect a sophisticated mechanically flexible separation unit was designed and manufactured in house. The movable fraction rack contains a permanently installed waste collection unit and two rows of fraction vials for uranium plutonium collection, respectively. In order to support the robotic movement of the rack into an accurate position, precision point click-ins were manufactured into the rack rail system.

The entire system is controlled by custom-made software from CertusSoft, Inc. The software is developed under the latest .NET platform (3.5) and is using an SQL server database for storing data, hardware settings and process related information. The software package consists of three components: two for configuring hardware settings and robot positions and the main component (Phoenix Management System), which is designed to configure the overall system, program multiple procedures for different sample types and finally to execute these procedures.

2.1 Robot Configuration Manager

The Robot Configuration Manager is the application used to teach locations, racks and gripper distances (Figure 3). The names used to store these locations will later be used by Phoenix Management System application. All data generated by the Robot Configuration Manager is stored in the SQL server database.

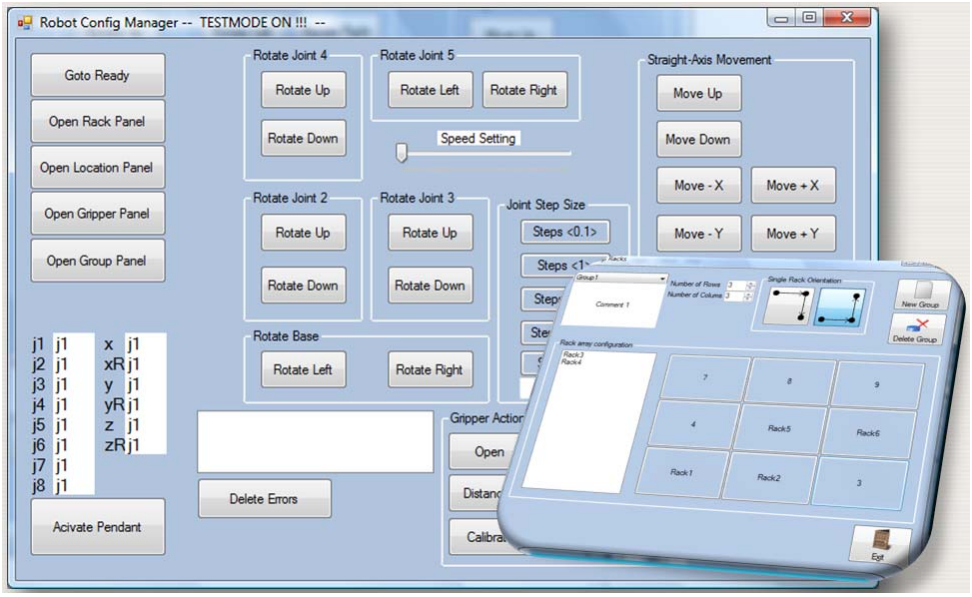

Figure 3. Example window from the Robot Configuration Manager.

The operator may choose to move the robot via the graphical user interface or via the teach pendent provided by Thermo/CRS. In either case the precise location is captured and stored in binary form by the application. Most tasks within the system require the robot to approach a location via two sub locations. These are referred to as safe/approach/base location in the software. The safe- and approach- location settings are optional, however they are normally turned off only in specific situations (stripping off pipette tips).

The software also provides a feature to define racks using three corner positions. In case of very large racks, experience has show that precision positioning can be difficult, if not impossible. Therefore the program allows for the introduction of so-called sub-racks.

2.2 ADAM Configuration Manager

The ADAM application handles all configuration settings for the ADAM 5000/TCP controller and all the modules used to control hardware such as pumps, IOs for drop counters and hotplate temperature.

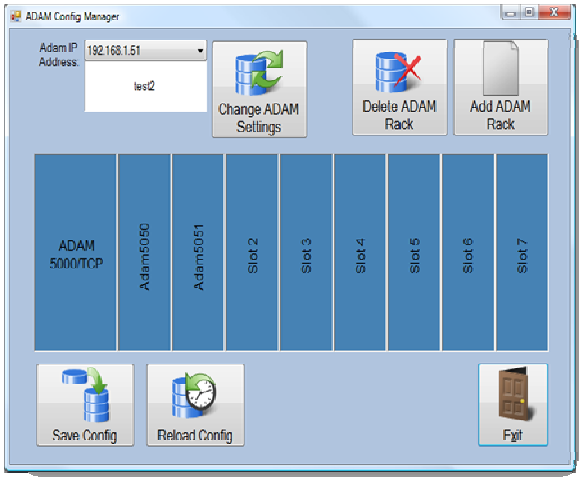

Figure 4. Example window from the ADAM configuration manager/

The configuration is stored in the SQL server database so the main application can easily access the information. Communication with the ADAM controller is handled over a TCP network connection.

2.3 Phoenix Management System

This application is an expert system, which simplifies the operator's tasks in setting up automated work tasks for the programmable robot arm, as well as for other components like digital IOs, pumps, barcode readers and others (Figure 5).

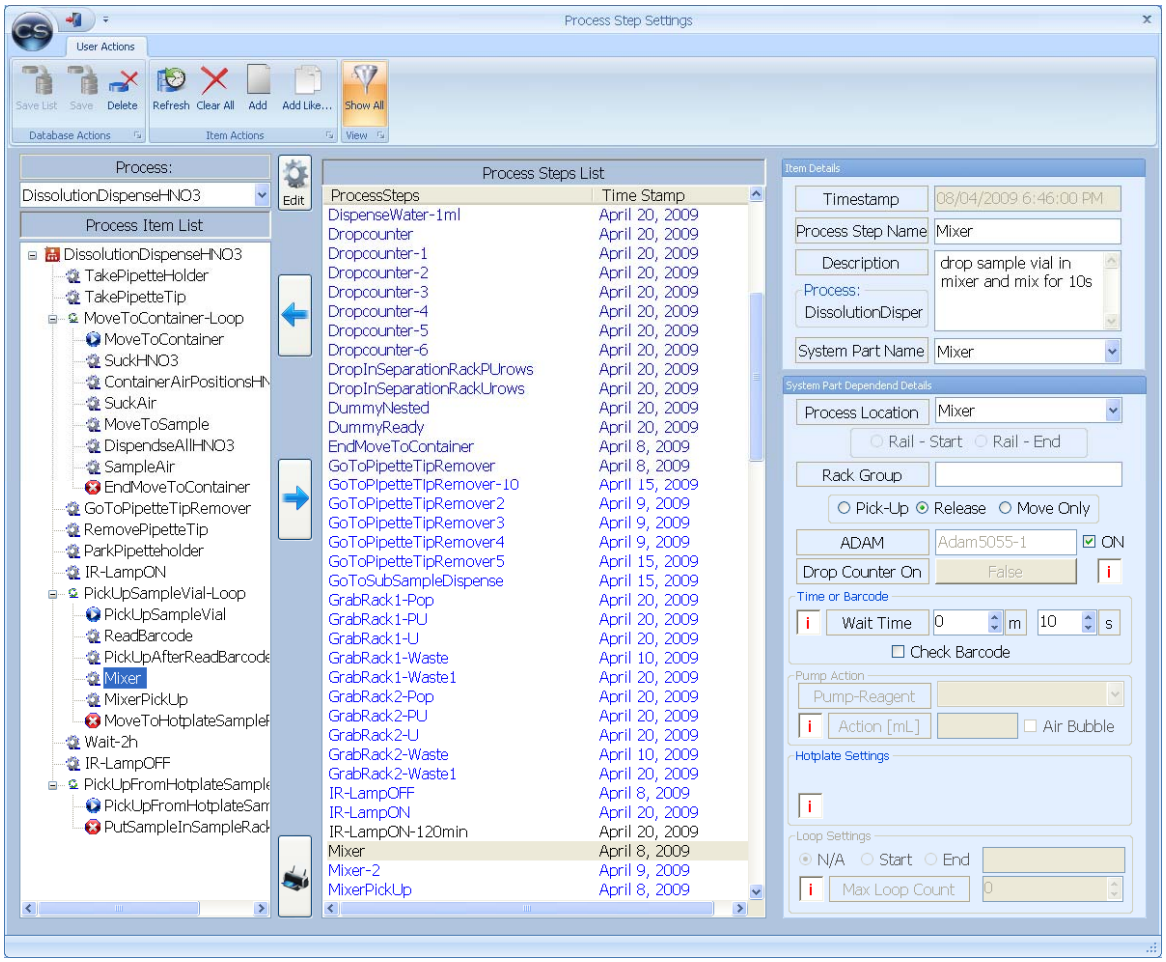

Figure 5. Window view of the Phoenix management system application.

This system was developed to allow a nonprogrammer to define all steps involved in a chemical separation procedure. The program assists the operator in providing check lists for the procedure to be executed before each run. Additionally, the program tracks the progress of a run so the operator can decide whether or not the system should continue a run after an unintended interruption (e.g. power outage). Changes in the chemical separation procedure or the box layout can be easily accommodated by the system. All data is stored in the SQL server database.

3. Adaptation to Glove Box Operation

A standard glove box was modified to accommodate the automated system and its components. Since the piston pumps are located outside the glove box, a sophisticated tubing system was custom built in house such that the tubing is introduced via sealed entry points at the ceiling of the glove box. Further sealed entry points were introduced into one of the side walls of the glove box to allow communication cables to reach the robot and the electric components inside the glove box. All electronic controllers are located outside the glove box to avoid corrosion.

In order to mitigate corrosion, a stronger air flow throughout the glove box volume was introduced. As mentioned above an additional separate exhaust vent is used for the heating casing. Prior

experiences show that these measures lead to an acceptable level of corrosion with minimum maintenance requirements.

Since a moving robot arm could harm equipment or operator, an emergency stop button was introduced to meet safety concerns. Failure of hotplate controllers is mitigated by the installation of a second thermo element with a separate hotplate controller, which provides an automatic emergency shutdown of the entire system in case of overheating. The same safety feature was applied to a smoke detector.

5. Conclusion

In order to reduce the radioactive dose rates from spent fuel and high-active liquid waste samples, and to increase the reliability of the chemical processing of these samples, a new five-axis robot arm and combined glove box has been developed in SAL. The system utilizes a state-of-the-art robot arm with a custom written software application for easy setup and control. Special consideration was made to assure that the application is easy to learn and modify by a nonprogrammer. The glove box is designed to take advantage of the improved reach and mobility of the robot arm, and to mitigate corrosion caused by build up of acid fumes. Once the system is implemented, it will be capable of routinely processing the chemical separation of up to 32 spent fuel samples per 24 hour period, resulting in a sample throughput gain of 50% to 100% over the manual processing approach.

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Reading the information inherent to uranium materials

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Abstract:

Uranium materials hold more information than normally exploited for safeguards evaluation purposes. The information inherent to the material arises from the isotopic, physical or chemical properties of the sample. Retrieving such additional information is of growing importance in nuclear safeguards, and of fundamental importance to nuclear forensics and environmental sciences. For example, the measurements of minor isotope abundance ratios of uranium in micrometer sized particles collected at nuclear facilities may be used to (1) substantiate equipment or plant design modifications, (2) indicate information about irradiation history, or (3) evaluate mixing, and decay scenarios. Furthermore the isotopic information inherent to uranium materials is also of relevance in geochemistry and earth sciences. Uranium series dating is one such tool that can be applied in such different fields as climate change, seismic hazard, archaeology, and volcanology. The chemical impurities in uranium samples may be native to the ore or process-inherited. In nuclear forensics, the chemical impurities may provide a wealth of information on the provenance of the material and the process it was subjected to. Reliable measurements of trace element concentrations and minor isotope ratios in uranium and the interpretation of the data are the basis for exploiting this source of information. With regard to the minor abundance ratios of uranium and impurities in uranium materials, there is a continuing effort invested by safeguards authorities, manufacturers, research institutes and analytical laboratories in improved measurement techniques, procedures, quality control and reference materials. These two specific topics, measurement of minor isotopes and chemical impurities in uranium materials, were addressed in two dedicated workshops organised by the ESARDA Working Group on Standards and Techniques for Destructive Analysis (WG DA) together with the IAEA. Participants in these workshops included representatives from the main European and international nuclear safeguards organisations and nuclear measurement laboratories, as well as fuel manufacturers and experts from geochemistry and environmental sciences institutes. This paper provides a synthesis of the workshop recommendations on protocols for sample collection, sample preparation, analytical methods, data evaluation techniques, quality control and reference materials for measurements of minor isotopes and impurities in uranium samples. Performance recommendations are also proposed, complementing the current set of International Target Values for Measurement Uncertainties of Nuclear Material for Safeguards.

Keywords: Minor isotopes of uranium, chemical impurities, nuclear safeguards, nuclear forensics, international target values

1. Introduction

The IAEA is currently in the 20/20 vision for the future re-defining its role for a safe, secure and peaceful use of nuclear science and technology [1,2]. Two of the 21st century's greatest challenges are to satisfy increasing energy demands and climate change; implicit in these challenges is the growing role of nuclear power. Effective IAEA safeguards contribute to international peace and security. The IAEA's role towards nuclear disarmament has been emphasised again after US-President Obama recently declared the United States government's commitment to re-invigorate nuclear arms control and nonproliferation as an international priority on its political agenda [3, 4]. Safeguards authorities have to meet these new challenges and thus require a strong verification and detection system in place to safeguard nuclear materials and activities for peaceful purposes, to ensure that there is no diversion of nuclear material to military use. To this end the IAEA and European Commission pledged

to work together across several fields related to the peaceful applications of nuclear energy and issued in May 2008 a statement on Reinforcing Cooperation on Nuclear Energy for Peace and Development. Support to the IAEA for safeguards research and development through the Joint Research Centre – JRC is seen as one of the outstanding examples of the effectiveness of this cooperation [5]. Under the Additional Protocol, the IAEA has responsibility to verify the completeness of a State's declaration [6]. On different stages of the nuclear fuel cycle - from mining to enrichment to nuclear fuel fabrication - inspectors take samples and draw conclusions on the origin of nuclear material, the process used and the feed material. Export and Import controls provide the IAEA with the means to determine the origin of material. A significant increase in uranium deposit processing is expected to require a similar increase in the workload in verification of material origin. The future verification regime cannot rely only on traditional measurements of uranium concentration and enrichment alone. Or as Olli Heinonen, the Deputy Director General, Head of the Department of Safeguards, pointed out - *we cannot meet tomorrow's challenges with yesterday's tools.*
Therefore safeguards authorities have already recognized the fundamental importance of measurements of minor uranium isotopes in environmental sampling. Nowadays the minor uranium ratios are measured in almost all of the environmental samples taken by inspectors. Furthermore chemical impurities are considered a potentially useful additional source of information. In nuclear forensics similar questions have to be answered by the investigators in view of source attribution of a seized uranium material. The complete isotopic information inherent to uranium materials is also of relevance for making age determinations of geological samples for various earth sciences applications [7].

2. ESARDA WG DA workshops

Uranium materials hold more relevant information than what is routinely examined for safeguards purposes. To include this information could be very beneficial towards a strengthened safeguard regime for the future. To address this topic the *ESARDA Working Group on Standards and Techniques for Destructive Analysis* (WG DA) dedicated in April 2008 a workshop to the *MEASUREMENTS OF MINOR ISOTOPES IN URANIUM IN BULK AND PARTICLE SAMPLES* and another one in March 2009 to *MEASUREMENTS OF IMPURITIES IN URANIUM SAMPLES* [7, 8]. The first was hosted by the Institute of Reference Materials and Measurements (IRMM) in Geel, the latter by the Institute for Transuranium Elements (ITU) in Karlsruhe. The first workshop day was dedicated to presentations from workshop participants in one plenary followed by topical sessions. The findings and points of discussions of this first workshop day were addressed in more detail during the second workshop day in separate working groups dealing with the required measurement quality, measurement techniques/protocols and data evaluation for the analysis of bulk and particle sample for minor isotopes in uranium and for the identification of characteristic patterns or parameters in impurity analysis. Finally recommendations for advances and further research work to access and evaluate the information inherent to uranium materials were given by the workshop participants. The intention was to exchange views and information on the needs, the applicable measurement techniques and the required quality of measurement results and statistical evaluation techniques. All in all, about 80 participants from 30 institutes attended the workshops, which is quite remarkable for workshops on such specific topics. Participants came from Europe, the US, Australia and Asia, including the main European and international nuclear safeguards organisations, nuclear measurement laboratories, fuel manufactories but also experts from geochemistry and environmental sciences institutes. The positive response was a confirmation that the workshop organisers indeed chose two topics that are of great interest to a broad community.

2.1 Objectives

The workshop objectives were to identify the needs for uranium minor isotope and chemical impurity measurements in nuclear safeguards and related areas, to review the current state-of-the-art of relevant measurement techniques, use of impurity patterns for evaluation purposes in different area and to increase the knowledge exchange between nuclear safeguards, industry and environmental sciences.

The workshop objectives were the following:

- To identify the needs for uranium minor isotope measurements in nuclear safeguards and related areas
- To formulate the needs and requirements related to chemical impurities in uranium of Safeguards Authorities, fuel manufacturers and nuclear forensics laboratories
- To review the current state-of-the-practice / state-of-the-art of relevant measurement techniques and to identify measurement challenges that should be addressed, including quantification, limit of detection, calibration, standards and reference material, quality control, sample preparation
- To Identify evaluation challenges such as statistical tools, reference data, comparison samples, data interpretation and attribution of parameters to source material or to process type
- To increase the knowledge exchange between nuclear safeguards, nuclear industry and environmental sciences
- To draft recommendations for
    - Improved Measurement Techniques and Procedures for minor uranium isotope ratios
    - Further research work, for the identification of characteristic parameters
    - Quality Control and Reference Materials
    - Establishing performance goals
    - Data Interpretation
    - New fields of application

2. Minor isotopes in uranium samples

High-Quality Measurements of minor isotopes in uranium are of fundamental importance in nuclear safeguards, nuclear forensics and earth sciences. The information on the isotopic composition of uranium samples taken at or close to nuclear facilities, discovered in illicit trafficking or investigated by geochemists is essential when drawing conclusions on the history of a material.

2.1. Environmental sampling

The minor uranium ratios are measured in almost all of the environmental samples. The analysis is carried out at the Safeguards Analytical Laboratory (SAL) within the IAEA a Network of Analytical Laboratories (NWAL).

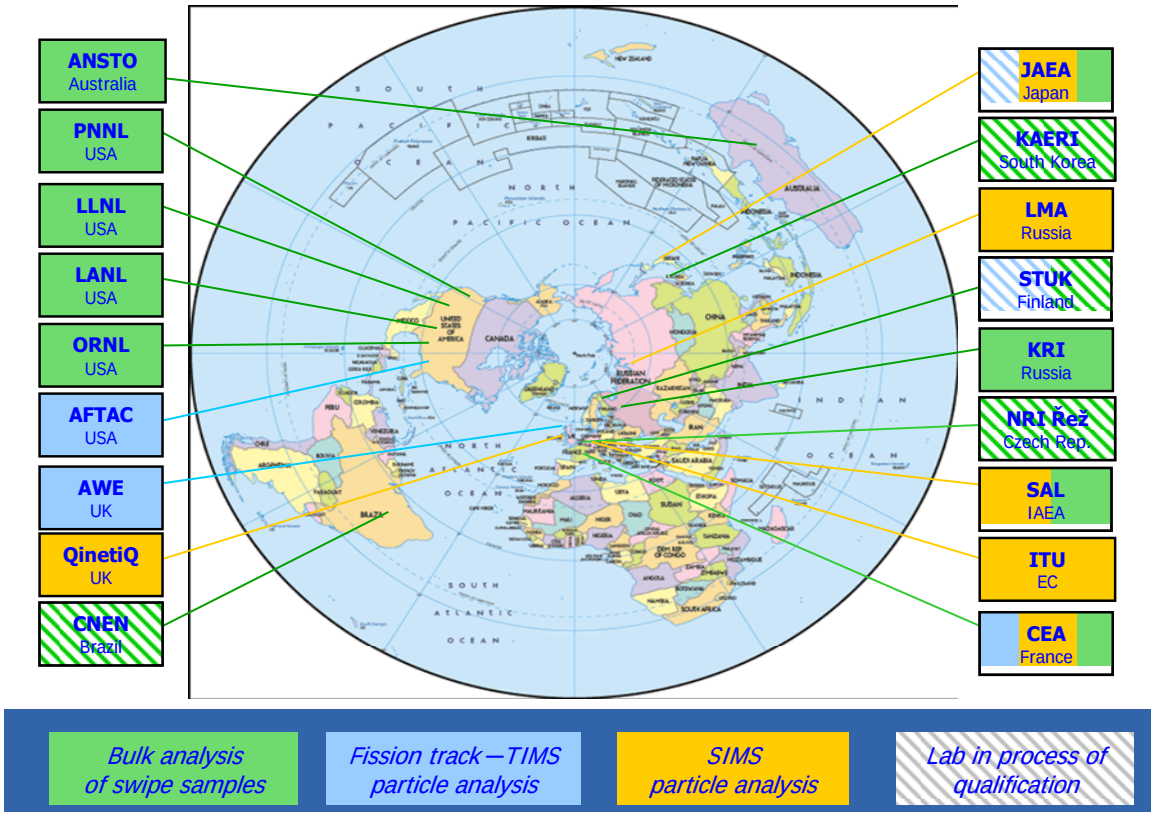

Figure 1: IAEA Network of Analytical Laboratories

Measurements of minor isotopes in swipe samples taken at enrichment plants indicate different feed materials or enrichment processes, and may provide additional information about equipment or plant design. Minor uranium isotopes from swipe samples taken from hot cells may reveal information about irradiation history, and also may clarify mixing and decay scenarios. The interpretation of environmental sampling results is often challenged by insufficient accuracy in the minor isotope ratios of uranium, such as for samples only slightly differing from natural uranium. In addition, measurement and subsequent evaluation of samples where ^{236}U is actually present at low abundances (i.e. 1-200 ppm) is very challenging. Therefore, improved measurements of uranium minor isotopes, including a low detection limit for ^{236}U and reliable uncertainty estimates, are desirable and crucial for the evaluation. The current safeguards challenge is to provide measurement results in environmental samples of the major ratio $^{235}U/^{238}U$ with a relative standard uncertainty of ± 1 %. The NWAL laboratories can report an upper limit in case the isotope amount fraction of the minor isotopes is below 1ppm, particularly for $n(^{236}U)/n(U) < 1ppm$. At present the IAEA is discussing the need for performance requirements for the measurements of major and minor ratios in uranium samples, and what can be considered as good practice for NWAL laboratories providing reliable high quality measurement results.

2.2. Nuclear forensics

Nuclear forensics aims to identify the origin and intended use of the material. Different parameters such as macroscopic appearance, microstructure, isotopic composition, elemental composition, impurities and decay products need to be looked at in a material to succeed in source attribution and age determination. Different analytical methods from safeguards, material sciences and geology are used to achieve this goal. Particularly minor isotope measurements in uranium samples help in source attribution and are an important part of 'nuclear fingerprinting' of discovered, unknown material. As an example the minor isotope measurement results from yellow cake samples from Iraq (1991) are shown in Figure 2. The difference in the minor isotope ratios revealed that not all uranium samples originated from declared material.

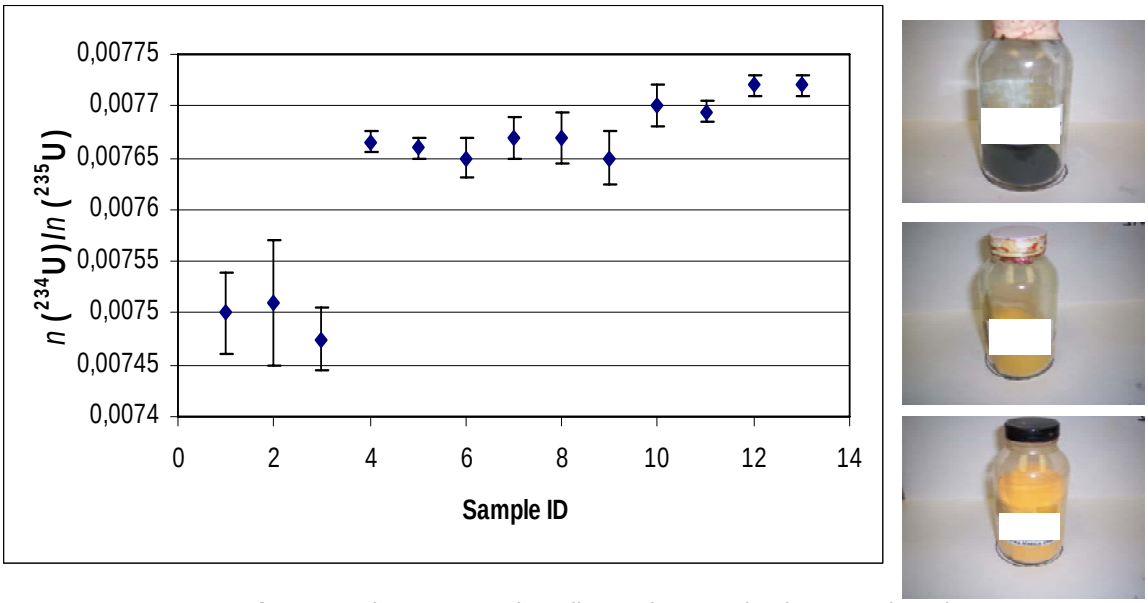

Figure 2: Minor Isotopes in Yellow Cake Samples from Iraq (1991)

Relations between materials can be established by means of minor isotope measurements in uranium samples (see Figure 3). The need for low detection limits for ^{236}U -, for lower uncertainties for ^{234}U - and to include ^{232}U measurements in the considerations is of major importance in nuclear forensics. Due to the fact that analytical results in nuclear forensics have to stand up in a court of law, the reliability and comparability of measurement results of minor isotope ratios in uranium samples need to be guaranteed and monitored via the correct use of reference materials and quality tools.

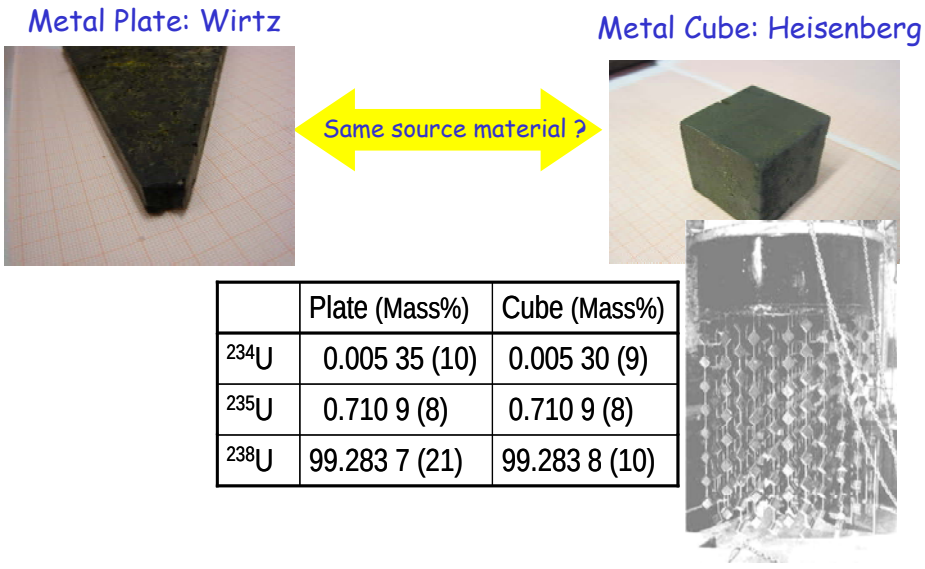

	Plate (Mass%)	Cube (Mass%)
^{234}U	0.005 35 (10)	0.005 30 (9)
^{235}U	0.710 9 (8)	0.710 9 (8)
^{238}U	99.283 7 (21)	99.283 8 (10)

Figure 3: Metal cube originating from the German nuclear energy project world-war II

2.3. Geochemistry and Earth sciences

Uranium series dating is a powerful tool that is applied in different fields, such as climate change, seismic hazard assessment, archaeology, and volcanology. A uranium-series date is not actually a true age but the time since uranium was chemically fractionated from its daughter element, e.g., dissolution in water, where protactinium or thorium daughters are insoluble. This method requires very accurate measurements of the uranium isotopic ratios as well as the uranium/daughter ratio, and applies over an age range from recent to 350 000 years. For example bones have a large capacity to adsorb uranium. The diffusion-adsorption model for U-series was developed to predicts the growth of ^{234}U and ^{230}Th across a bone section [9]. A number of samples across the bone should give a consistent, robust age. Because precise dates require high accuracy isotopic and elemental ratio measurements for the actinides, there is a continuing strong effort invested in reliable isotope ratio measurements of uranium and thorium for earth and environmental sciences applications.

3. Chemical impurities in uranium

Trace element characteristics are considered the most important tool for defining the origin and underlying processing of uranium material. In nuclear safeguards and nuclear forensics, the aim is to draw reliable conclusions on the source of uranium materials, the process it was subject to, the age of the sample and the purpose it was intended for.

3.1. Nuclear safeguards

Chemical impurities are a useful additional source of information complementary to traditional measurements of uranium concentration and enrichment. Metallic impurities may be inherent (ore itself) or adherent (Process /Storage) to a sample. Therefore it is of fundamental importance to the IAEA and the scientific community to investigate the behaviour of impurities during processes such as conversion. Source uranium ore concentrate (UOC) are characterised to verify compliance with source declarations, yellow cake samples are analysed to monitor the ore supply purchased and processed in the plant. Caution is required in sampling, sample preparation and sample introduction to avoid contamination.

Table 1 gives a list of metallic impurities of interest to the IAEA in uranium Materials.

Construction Material Elements		Nuclear Elements	Other Elements		No Interest	
Al	Nb	B	Ca	Ba *	Rb	Ag
Ni	Co	Be	K	As *	Cs	Au
Cr	Si	Cd	Na	Sb *	Ra	Hg
Cu	Ti	Ce	P		Sc	Ga
Fe	V	Gd	Pb		Ac	In
Mg	W	Hf	Sn		Tc	Ge
Mo	Zn	Li	Sr		Re	Bi
	Mn	Ta	Th		Ru	Se
	Zr	Y	REE		Os	Te
					Rh	Po
			* *only in Yellowcake*		Ir	
					Pd	
					Pt	

Table 1: Metallic impurities of interest to nuclear safeguards

In a recent study it was shown that characteristic patterns on uranium ore type can be derived from the combination of the information gained via isotope ratio measurements and from impurity analysis Lead and strontium isotopes proved particularly useful in classifying UOCs, because the lead composition was dependent upon the age of uranium, whilst strontium varied considerably with location [10]. Currently, although three NWAL are supplied samples for impurity determination, the IAEA is actively seeking to qualify more labs. Those laboratories must have comparable performance criteria, and emphasis was placed on the need for consistency in limits of detection to enable a straightforward data evaluation process. IAEA expects a constant supply of about 100 U samples per year for impurity work.

3.2. Nuclear forensics & illicit trafficking

Impurities measured from seized uranium samples may be from source material, or can come from intentional additions during processing, such as poisons (gadolinium, erbium) or alloying components (e.g. aluminium), or from accidental additions, resulting from cross contamination.
The challenge is to enable sample attribution to an existing data set and to compare different data sets. Currently studies are carried out to look at rare-earth patterns of about 100 uranium ore concentrates to identify which impurity comes from which source and how the process would affect impurities [11]. Furthermore ITU is investigating whether anion measurement in uranium ore concentrates by ion chromatography focussing on sulphate, phosphate and halide concentrations can provide information suitable for nuclear forensics applications [12]. Interstitial elements - carbon, hydrogen, nitrogen and sulphur – may provide some indication of how the uranium was produced, for example, high interstitial carbon content is indicative of preparation in a graphite mould. For sample attribution it is also valuable to include the specifications for impurities supplied by manufacturers.

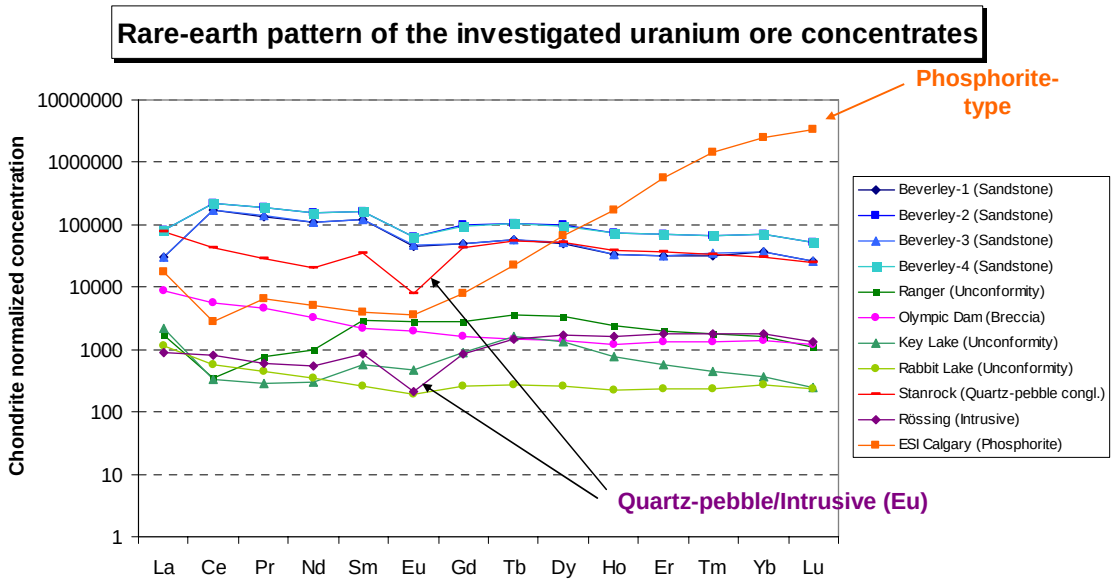

Figure 4: Rare-earth patterns in UOCs

4. Measurement approaches

State-of-practice and required advances in the relevant mass spectrometry techniques for the measurement of uranium isotope ratios and impurities were reviewed by the workshop participants.
The methods commonly used for the measurements of minor isotope ratios in particle analysis are Fission Track Thermal Ionisation Mass Spectrometry (FT-TIMS) and Secondary Ion Mass Spectrometry (SIMS). Recently there have been also attempts to use Laser Ablation Inductively Coupled Plasma Mass Spectrometry (LA-ICP-MS) to analyse uranium particles. For bulk analysis TIMS, multi-collector ICP-MS and accelerator Mass Spectrometry (AMS), particularly for the small $^{236}U/^{238}U$ ratio are the applied measurement techniques. Impurity measurements are performed with

ICP-MS. Details on the different techniques and laboratory performance for minor isotope and impurity analysis are discussed elsewhere [7, 13]. Here we focus on the recommendations towards needs and advances in the future.

4.1. Bulk analysis of swipe samples

Environmental samples are taken to verify the absence of undeclared nuclear activities. Bulk analysis, of the U and/or Pu on a swipe sample requires chemical processing and the detection of small signals using primarily thermal ionization or inductively coupled plasma mass spectrometry. The results are often important for evaluation of isotopic mixing scenarios. In some of these cases, a check for possible material matches in the IAEA databases is requested, based on ^{234}U, ^{235}U ^{236}U abundance. One common aspect for all mass spectrometric techniques is the linearity response of the detectors, in particular for Secondary Electron Multipliers (SEM) in view of proper linearity investigation and required corrections. It was agreed to share observations and developments for SEM detectors within the ESARDA WG DA. Another detector aspect for the accuracy of minor ratio measurements is the peak tailing effect. In order to overcome the problem of peak tailing during minor U ratio measurements, the so-called Modified Total Evaporation technique (MTE) was developed at NBL and refined at IRMM [14, 15]. This method permits the routine determination of the ^{235}U/^{238}U major ratio as well as of the ^{234}U/^{238}U and ^{236}U/^{238}U minor ratios within the same measurement, with improved precision and accuracy. Currently the MTE is implemented at SAL for nuclear measurements. For many applications ^{236}U detection limits (< 0.1 ppm desirable) and the ability to detect deviation from a 'natural' ratio of about 55 ppm, are driven by the presence of isobaric interferences, as well as blank levels. Isobaric interferences are observed in all types of isotope mass spectrometers. For ICP-MS, the hydride correction can be significant, i.e., from ^{235}U+H at mass 236. Some interferences on the ICP-MS can be eliminated by using high mass resolution or by a desolvating sample introduction.

4.2. Particle analysis of swipe samples

The need was emphasized for highly precise and accurate minor and major uranium isotope measurements of individual uranium particles taken from swipe samples, as well as sufficiently low detection limits for the minor isotope ratios, in particular for ^{236}U (<1ppm). Based on the ^{233}U, ^{234}U, ^{235}U, and/or ^{236}U content, swipes taken at enrichment plants indicate different feed materials, enrichment processes, and may provide additional information about equipment or plant design, operational parameters, etc. Swipe samples taken at facilities with hot cells indicate key information about irradiation history, and also help evaluate mixing, and decay scenarios. FT-TIMS presently provides the highest quality minor U isotope data for individual particles, however this technique is limited to only a few laboratories due to the specialized requirements and high cost. SIMS is the most widely applied technique for swipe samples analysis, however it is limited in its performance due to its problems caused by unresolvable isobaric interferences. The overall analytical uncertainty, in particular for SIMS results, is often under-estimated. Measurement bias is often seen in the minor isotope results from particle analysis with SIMS which may make evaluation of inspection results difficult. A new candidate for particle analysis is Large Geometry (LG)–SIMS, which promises to have FT-TIMS quality for the minor U isotopes. LA-MC-ICP-MS is a technique under development that should be further investigated, showing also a potential application in particle analysis. Another important aspect is the selection of the particles of interest. Scanning Electron Microscopy was mentioned as a tool for finding particles of interest and also as a tool for SIMS efficiency measurements by taking microscope pictures before and after the measurements.

4.3. Impurities

Trace element analysis of nuclear material samples is accomplished primarily using ICP-MS techniques. ICP-MS is a versatile and sensitive technique, capable of measurement of virtually all but the lightest elements of the periodic table. Currently, the IAEA uses the NWAL for impurity determination, however a new ICP-MS laboratory at SAL will soon be in routine operation for making impurity determinations. Sample dissolution procedures are being tested whilst measurement and quality control procedures are under development. At high mass number, interferences are inevitable and need to be corrected for with external calibration, using matrix-matched standards. The role of minor isotopes and transuranics in determining origin was pointed out, for example ^{230}Th/^{238}U which

should be $<10^{-10}$ in freshly-separated uranium. In fact, by ICP-MS, the actual measured ratio can vary by orders of magnitude as a consequence of peak tailing. This problem is now addressed by MS-ICPMS instrument manufacturers, and energy filters are also available. Besides ICP-MS also other techniques still play a role in analysis, depending upon the elemental signatures under investigation. For those trace elements that were relatively high in abundance, XRF was particularly useful due to its non-destructive assay. ICP-OES and atomic absorption (AAS) have been largely replaced by ICP-MS, but still play a valuable role in the measurement of uranium without dilution.

5. Identification of characteristic patterns

The use of isotope ratio measurements for source and process identification requires high precision measurements of the enrichment $^{235}U/^{238}U$ and the minor ratios $^{234}U/^{238}U$, $^{236}U/^{238}U$. Based on the full isotopics of uranium material conclusions can be drawn on possible origin, production history, nuclear fuel history and differentiation between presences of small amounts of anthropogenic uranium in the environment. Minor isotope analysis is particularly attractive for monitoring the conversion to UF_6 where non-volatile impurities are removed, and any remaining impurities are enriched, whilst pool-feeding is utilised at enrichment plants. As a consequence, it is not possible to trace the origin of material through enrichment via impurity analysis. ^{234}U concentrations in the feed material can yield information on the origin, ^{232}U, ^{236}U on the burn-up. ^{234}U concentrations in the product material can yield information on down blending or re-enriched tails, ^{232}U, ^{236}U on reprocessed uranium. Particularly the need for low detection limits for ^{236}U -, for lower uncertainties for ^{234}U - and to include ^{232}U measurements in the considerations being of major importance in nuclear forensics.

Identification of a uranium material always includes a combination of macroscopic appearance, microstructure, isotopic composition, elemental composition and impurities. Rare earth patterns in addition with anions and cations measurements and interstisital elements are also of interest in some cases. Recently IRMM concluded a study on the determination of fluorine in uranium oxyfluoride particles as an indicator of particle age [16].

5.1. Data evaluation

Statistical tools play an important role in data evaluation, enabling quantitative comparison between samples. Groups of samples should first be identified that show clear statistical differences. Multivariate statistical analysis was recommended, e.g. cluster analysis or principal component analysis, with a confidence level determined as to whether a particular sample belonged to a group. There are existing IAEA and other databases where a result can be quantitatively compared. Parameters should be prioritised, with the most important ones evaluated first, such as isotopic composition, metallic impurities and rare earth patterns. Particularly unification of the way in which detection limits are expressed in these databases is crucial also uncertainties could be considered. The detection limits and analytical uncertainties of the methods used to make the measurements should be clearly stated by each laboratory. To assist in establishing minimal requirements for detection limits and uncertainties, current state of practice methods should be assessed via interlaboratroy comparisons. It was re-iterated that proper uncertainty estimation is important for reporting of results for minor uranium ratios and impurity measurements. The GUM approach for uncertainty estimation may be one way of establishing a baseline among laboratories. An appreciation of uncertainties is required within the evaluation process, in order to assess the significance of a measured value and the weight that might be given to a result within the evaluation. Knowledge of the process and chemistry of the impurities is essential, and should be used to determine the optimum means of presenting data for statistical analysis. However, combination with scientific information and input from industry is essential in order to enable a proper assessment. For example, simply the colour of certain materials may give a strong indication of their origin. Such non-numerical information is difficult to incorporate into a statistical assessment, but relatively easy to gather and should be fed back into the evaluation. Overall, subject to constraints of confidentiality, it was considered that greater interaction between the laboratory and authority, as it is already the case in nuclear forensics, would both reduce the analytical effort required and assist in the timely and accurate attribution of materials [13].

6. Reference materials and quality tools

Any technique used must be validated with appropriate quality assurance/quality control oversight. For all methods appropriate reference materials are needed. To validate the technique's performance, real samples are required. However, no "real" standards are available for impurity analysis, suggesting a need for the preparation of standards and interlaboratory measurement and certification. Interlaboratory comparisons (ILCs) were proposed to assess measurement procedures through analysis of an uranyl nitrate solution and subsequently involving the analysis of solid samples within a uranium matrix (e.g. uranium oxide). This would enable laboratories additionally to test their dissolution procedure and enable comparative measurements of trace elements and minor isotopes. The availability of appropriate reference materials was recognised as being of critical importance. The production of a certified "Trace Elements in Yellowcake" standard was recommended. Impurities certified should include rare earth elements, transition metals, calcium, sodium and potassium. CETAMA provides uranium materials certified for impurities [17].

In environmental sampling the need for uranium particle Quality Control and Reference Materials was expressed. Mainly, uranium reference particles U_3O_8 or $UO_2$and UF_4 with 1-2 pg per particle (certified amount for efficiency measurements) with isotopic compositions of natural uranium, low-enriched uranium, and highly enriched uranium. As a response to this need IRMM coordinated the first NUSIMEP interlaboratory comparison on uranium isotope amount ratios in uranium particles. NUSIMEP-6 was oragnsied for all laboratories dealing with analysis of uranium in particles, in particular for the IAEA network of analytical laboratories (NWAL) for environmental sampling [18].

For bulk analysis, it was mentioned that reference materials (RMs) with adequate uncertainties for minor ratios should be used on a routine basis. Recently, IRMM has re-certified the IRMM-183-187 series for the minor ratios, which were found to be very suitable reference materials for this purpose [19]. This series has now been adopted by SAL as well.

7. Recommendations

Many of the technical challenges in measuring the minor uranium isotopes and impurities in uranium materials are shared by safeguards, nuclear forensics, industry and earth sciences communities, and advances would benefit them all.

General recommendations:

- Production of well certified reference materials for or trace elements in yellowcake and isotopic composition and amount content of uranium particles
- Organisation of Interlaboratory comparisons in the field of particle analysis and impurity measurements of uranyl nitrate and uranium oxide samples
- Uncertainty estimation has to be carried out according to the *Guide to the expression of uncertainty in measurement (GUM)* [20]
- Further consideration should be given to the need for performance standards (International Target Values) in minor isotope ratio and impurity measurements to support safeguards;

Recommendations particularly for measurements of minor isotopes in uranium

- The latest developments in measurement protocols should be included in commercial instrument software; a regular exchange of software issues among the ESARDA WG DA members
- Careful detector calibration is required and proper identification of interferences
- New methods, like LA-ICP-MS should be investigated further and mature applications have to be developed

Recommendations particularly for measurements of impurities in uranium

- The ESARDA DA working group should coordinate the preparation of best practice documents for impurity measurements in uranium, based upon submissions from participating laboratories
- Uncertainties and detection limits must be reported, with a unified approach to the expression of detection limits
- The ESARDA DA working group should explore opportunities to work with industry and share expertise, data and samples appropriate to measurement of impurities in uranium
- Further research should be undertaken in data pre-processing, evaluating the optimum parameters for measurement and their treatment;
- A comparison should be undertaken of different statistical tools using the same data set

8. Conclusions

Organising these workshops with a clearly defined technical focus showed that 'reading' the information inherent to uranium materials correctly conclusions on history and processes can be drawn in nuclear safeguards, nuclear forensics and geochemistry. Although there are limitations in measurement techniques and data evaluation, a combination of physical, chemical and isotopic patterns/properties of U inspection samples is valuable for assessing the completeness and correctness of state delcarations, and for attribution of seized nuclear materials. Further research in this field is definitely valuable to strengthen European and international safeguards systems. In particular, the participants from environmental sciences, industry and nuclear safeguards would like to continue to discuss/exchange information also after the workshop, using the ESARDA WG DA as platform to share the outcome of these workshops with a broader community.
The conclusions from both workshops will also feed into the discussions on the revision of the International Target Values (ITVs) [21]. Particularly the ESARDA WG DA will liaise with the IAEA. A meeting on the ITV revision is scheduled at the IAEA in Vienna to finalise and release ITV version 2010. Emphasise will be given to the fact that ITV 2010 has the potential to be a guidance document for the non-nuclear measurement community.
The response to these workshops exceeded the expectations of the organisers with respect to participation and to meeting the objectives. This is also a confirmation that the work of the WG DA is beneficial and appreciated by a wide community.

9. Acknowledgements

The authors very much acknowledge the contribution from all participants to the ESARDA WG DA workshops.

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SESSION 4

CONTAINMENT AND SURVEILLANCE – I

A Design and Simulation Tool for Nuclear Safeguards Surveillance Systems

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Keywords: nuclear safeguards, surveillance, virtual reality, design, planning

Abstract

This paper presents a preliminary virtual-reality based application for designing and simulating surveillance systems for nuclear safeguards. The primary purpose of this tool is to help the nuclear inspector in planning the installation of safeguards surveillance systems. In practice, this means the selection of the appropriate equipment and accessories and their location in the real environment under surveillance. The current version supports cameras-based surveillance systems. The proposed tool has the capability of simulating plant operations and of checking its interaction with the surveillance system during normal operations. Furthermore, the user can iteratively assess the vulnerability of the designed surveillance system by simulating diversion scenarios with moving items and characters, by changing its parameters and, if necessary, by introducing redundancy in the equipment. As a consequence, the nuclear inspector can reach a satisfactory design in terms of robustness and cost-effectiveness, and improve the efficiency of the on-site installation phase. The paper details on a case study the implemented tools for the three-stage process, which consists of virtual-reality environment modeling, surveillance system location & configuration, and simulation of normal and diversion scenarios.

Introduction

In the Nuclear Safeguards literature, an early reference [1] outlines possible uses of virtual-reality tools for training and planning to nuclear sites inspections. In particular, [2] reports a prototype system for training on complex equipment.

The aim of this virtual reality-based tool for designing and simulating nuclear safeguards surveillance systems – named the **Surveillance Simulation Tool** (**SST**) – is to provide nuclear inspectors with an intuitive and easy to use tool for efficient and interactive planning for the installation of surveillance systems in nuclear facilities. In practice, this means that, beforehand, the inspector could interactively select the suitable surveillance equipment and plan how and where to install it in the plant. Apart from improving the robustness of the surveillance system, this would make the inspector on-site work much more efficient and thereby reduce associated costs.

Another objective is to provide a training tool for nuclear inspectors on surveillance systems. The current SST version permits to configure generic features of surveillance cameras and can be considered as a useful tool in a training programme. This training concept can be further developed implementing specific vendor features of surveillance cameras. It is worth to note that with respect to already available 3D demonstration tools from vendors of surveillance cameras, the SST has the advantage to be vendor independent and to provide modelling features and the capability to graphically and interactively simulate the operation of a facility as described in this paper.

The presented SST has been developed by the Institute for the Protection and Security of the Citizen of the EC-Joint Research Centre in the framework of its R&D Nuclear Safeguards activities.

The SST Application

General description of SST

The SST application is a software tool developed under Virtools™ 4.0, which runs on Microsoft Windows environments. Basically, the aim of the tool is to set up and check the behaviour of a surveillance system in a designed or imported 3D environment in which animations are run simulating normal plant operations or diversion scenarios. In the next section, the design process and the main parts of the application are explained.

Design process in SST

There are three main iterated steps in the usage of this application as showed in Figure 1. The first step consists of setting up a virtual scene which is a realistic representation of the real environment to be put under surveillance. The virtual environment has therefore to be modelled with appropriate dimensions and geometries and has to include relevant features and objects. With SST, the environment design can be achieved in various ways:

(i) a new scene can be created using the SST modelling tool,
(ii) a previously created and stored scene can be loaded into the application, or
(iii) a scene model created with 3D software modelling tool can be imported.

To make a realistic scene, SST offers a library of pre-defined objects - as walls, doors and windows - that can be added and intuitively positioned in the environment by clicking and dragging them. Once the scene is ready, the second step consists of setting up a *surveillance system* based on cameras. The SST application allows for the interactive loading and positioning of one or more cameras in the scene, and for the configuration of the individual camera parameters.

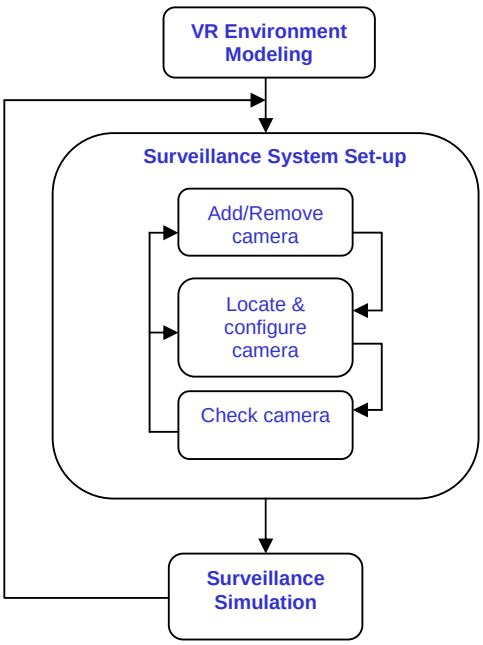

Figure 1. Scheme of the usage of the SST

Finally, the third step allows trying out the surveillance system with both static environment configurations and/or animated scenarios in the environment simulating normal plant operation or diversion cases. Afterwards, the obtained surveillance system configuration can be used to cost-effectively plan the on-site installation.

Environment Modelling Features

The SST application has two built-in environment models with a realistic look that can be loaded by the user: the Gorleben Nuclear Storage facility in Germany and the PERLA laboratory at the EC-Joint Research Centre Ispra site (Italy). These environments have been designed using 3DSmax and imported into Virtools. Otherwise, the user can create the representation of a new facility starting from a general rectangular room with configurable dimensions, including the width and height of perimeter walls. It is also possible to customize the look of the environment by changing the texture of the floor and walls. To fill the scene, the application includes a library of objects typically available in nuclear storage areas as casks and containers, but also office furniture and doors.

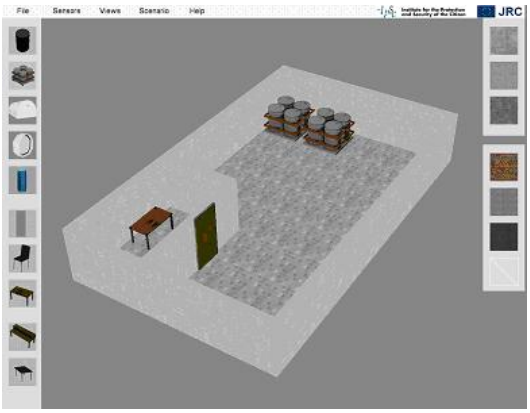

Figure 2. Custom Scenario created with SST

These objects can be positioned and rotated using the keyboard or mouse, with the collision detection mode activated to prevent objects crossing each other or crossing walls or the floor. The user can also add walls in the scene to divide it into several rooms. Walls properties as width, height, position and orientation, can be modified with context sensitive menus. Finally, characters can be added to the scenario and execute walking animations through keyboard commands. Figure 2 shows an example of the use of the SST tool creating a storage environment.

Finally, a distance measurement tool between vertexes is available to accurately position objects in the scene.

Camera Surveillance System Set-Up

There are 3 steps in the surveillance system set-up:

1. Loading cameras: up to four virtual surveillance cameras can be loaded inside the scene. Once a camera is loaded, a *camera view window* appears on the screen.

2. Camera position and orientation: by clicking on the bar of a camera view window, the camera-move mode is activated and a X-Y-Z frame (with red, green and blue arrows) attached to the camera is displayed; the user can click one of the X-Y-Z axes and drag the camera along it. Once the camera position is set, the camera orientation can be changed along one of the X-Y-Z axes by clicking the related axis with the middle mouse button. The camera view window is updated in real-time while the camera is moved; in addition, the camera coordinates and orientation are also updated in real-time on the camera set-up window when displayed (see Figure 4). Figure 3 shows an environment including two windows for cameras' view, and a frame attached to one camera.

3. Camera parameters configuration: A *camera set-up window* (see Figure 4) appears when clicking the camera set-up button on the camera view window bar. The camera set-up window displays camera's position and orientation, and allows the selection of the focal length, the CCD size and the resolution of the camera. The selection of these parameters updates the view on the corresponding camera view window.

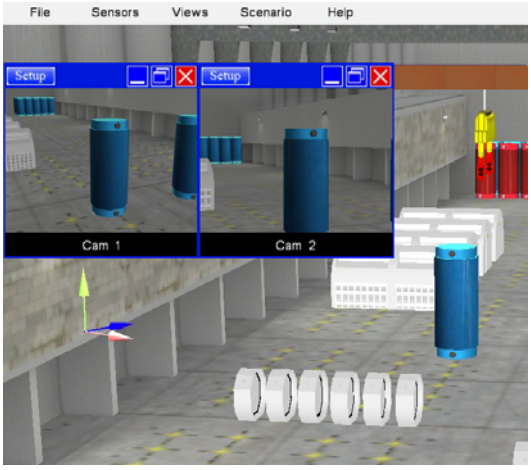

Figure 3. Camera view windows and frame

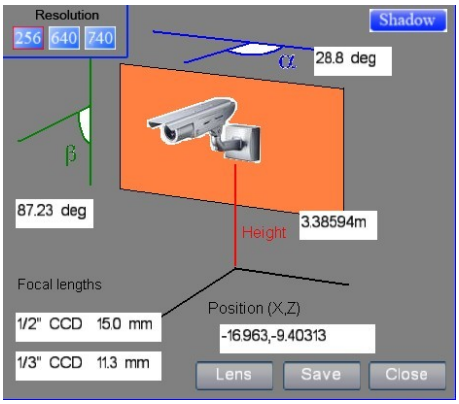

Figure 4. Camera set-up window

Focal length and CCD size can be selected in the *Lens* window, accessible from buttons on the *camera set-up window* menu (see Figure 5).

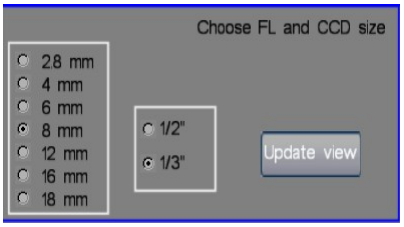

Figure 5 Lens parameter selection window

Surveillance Simulation

Analysis of blind areas

As the surveillance system is based on cameras, the SST has the capability to show which areas of the scene are not visible – also said 'blind' or 'shadowed' areas – for each surveillance camera.

Graphically, this is achieved by displaying a semi-transparent shadow generated by a focussed light "attached" to the camera while pointing in the same direction and with the same opening angle. Through a button in the camera set-up window, the user can assign a different shadow colour to each camera in order to distinguish blind areas from each individual surveillance camera. In addition, in order to improve scene clarity, the user can activate or deactivate the shadow on every object of the scene by a mouse click. When several cameras are pointing to the same object from different angles, shadows are superimposed and their intersection reveals the effective blind area for that specific configuration of cameras (see Figure 6). This is a practical way to check the redundancy of a surveillance system.

This capability of showing shadows is also available in animated scenarios with moving objects and it permits to analyze how blind areas change and may compromise the vulnerability of the area under surveillance.

Simulation of operations at a nuclear storage facility

The current SST version is tailored for nuclear storage areas and a typical operation in these sites is the relocation of spent fuel casks using bridge cranes. However, this software application could also be used for factories and plants with a similar flow of materials. For replicating a nuclear storage operation, the SST application has a tool for simulating crane operations as picking up, moving and releasing spent fuel casks in a location clicked by the user in the scene. In addition to surveillance camera views, the SST application also activates an orthographic view from the crane hook when the crane picks an object. Thereafter, during the simulated crane operation, the user can view how the position and shape of blind areas evolve and subsequently may add or modify cameras configuration and location to improve the surveillance robustness. The user can also verify that normal plant operations, like casks displacement, do not interfere with the area under camera surveillance. This process may be iterated until the user reaches a satisfactory configuration of the surveillance system.

Running diversion scenarios

The SST application allows for loading human characters, for programming animations by specifying their trajectory and the type of moves

(running, walking, etc.), and for synchronizing the animation with the simulated storage operation using a crane (as described in the previous section). This capability can be used for establishing diversion scenarios in which intruders trick the surveillance system taking advantage of the blind area from a casks moved by a crane.

Effect of camera resolution

Automated scene change detection (SCD) is an important feature of any video surveillance chain. It can be used at the front-end to reduce the images to be stored and/or transmitted to Safeguards authorities. When used at the back-end, it is normally used during the review process to focus inspectors' attention onto the scenes where activity is more likely to happen. Camera resolution, i.e., the inter-sample spacing at the object being surveyed, is a significant parameter that can dramatically influence the performance of a computer-assisted scene change detection. This is particularly true when the camera is aimed at large-sized areas. For example, as showed in figure 7, a moving character placed at the end of the storage area is simply too small (in terms of the number of image pixels in the image plane) for an effective detection of scene changes This happens as the camera field of view covers a wide area and/or the objects to be detected are just too far away. By using this SST capability, an inspector can simulate the effectiveness of a proposed camera system, including the selected optics, when associated to a computer-assisted scene change detection system.

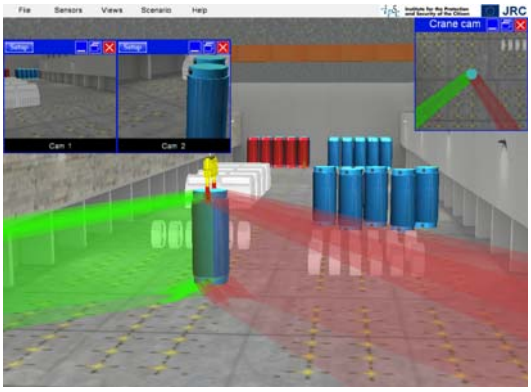

Figure 6. Shadows/blind areas with crane operation

Appraisal of the VIRTOOLS

As stated previously, the SST application tool has been developed under Virtools 4.0, a development platform for 3D real-time interactive applications. The advantages found in using Virtools are that it already includes many pre-programmed and useful behaviours which can be easily included in the final application. The final product can be created to be a stand-alone or web based application. Besides Virtools building functions, the final application can incorporate customized building blocks written in C++.

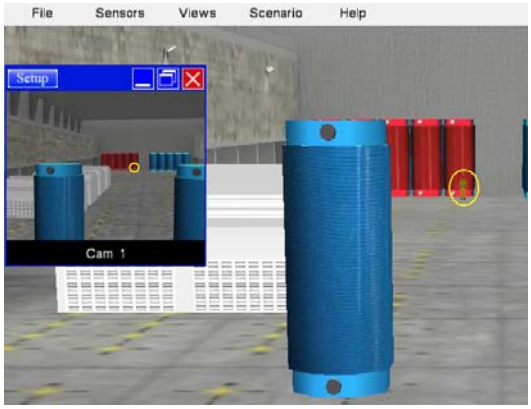

Figure 7. Resolution effect

Conclusions and Future Developments

The current version of the SST has been initially designed for storage areas of nuclear materials, but could easily be applied in any factory environment where goods are transferred and interact with the surveillance system.

Future extensions of the SST will cover radiation sensors and include the remote visualization of surveillance sensors, as demonstrated in the 3DPerla application [3], as well as the off-line review of surveillance data by nuclear inspectors.

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The Next Generation Surveillance System – Development Project Overview

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ABSTRACT

In late 2003, the International Atomic Energy Agency (IAEA) initiated the replacement of the aging Digital Camera Module 14 (DCM-14) based safeguards surveillance system with an international workshop to develop the user requirements for a new system. In early 2005, the development of the Next Generation Surveillance System (NGSS) was awarded to a German-U.S. development partnership as a four phase contract. To date, the first three phases (conceptual design, detailed design, and prototype development) have been completed and the zero production run units are being assembled. The final phase of the project includes the extensive testing as well as the vulnerability assessment necessary for authorization for routine use by the IAEA.

The following paper will outline the progress of the NGSS development project up to date; it represents the first comprehensive project overview presented to the non-proliferation community. It will begin by listing the essential user requirements that drove the conceptual design, especially the selection of system components and the implementation of the system architecture. It will then describe the detailed design and the early laboratory prototypes that allowed for further refinement of the system. Next, it will present the NGSS prototypes and the finalized NGSS architecture. An overview on the various testing steps and preliminary results available to date will follow. A brief summary of the NGSS fielding and DCM-14 phase out plan will conclude the paper.

INTRODUCTION

The concept of Continuity of Knowledge is a fundamental pillar upon which the International Atomic Energy Agency (IAEA) builds its inspection regime. After the initial physical inventory of nuclear materials has been verified at a safeguarded facility, subsequent safeguards measures are applied so that safeguards inspectors can draw the conclusion that no materials or technologies were diverted for weapons production or unknown purposes (usually a covert nuclear weapons program) during the inspection period. The first and most important safeguards measure is the inspector's access to the facility. Inspection activities include checking the material balancing accounts, taking samples for destructive analysis, and conducting measurements to verify materials quantities and compositions.

In between inspection visits, safeguards inspectors rely on a broad range of unattended instrumentation that is permanently or temporarily installed in the facility to maintain the Continuity of Knowledge. Among a variety of non-destructive assay instrumentation and containment measures, surveillance is one important activity to provide visual evidence of certain events of interest that can be analyzed by the inspector to see if they were part of routine facility operations or indicator of an anomaly or a diversion scenario. After using analog film surveillance systems and a first digital system in the 1980s and early 1990s, the Digital Camera Module 14 (DCM-14) was introduced for safeguards use in 1996. The switch to a digital data format allowed for the application of authentication and encryption schemes, thus making the data more reliable, and the automation of certain data processing functions. The later feature led to a significant expansion of the use of surveillance in safeguards, and today it is one of the fundamental safeguards measures to support Continuity of Knowledge.

After more than a decade of reliable service in a family of surveillance systems, the DCM-14 is nearing its end-of-life with critical components becoming obsolete while new technologies become available. In anticipation of this end-of-life stage, the IAEA initiated, in 2003, the replacement of the DCM-14 with a project called Next Generation Surveillance System (NGSS). After an initial period to define the user requirements of the new system, the development was finally contracted to a team of a German and a U.S. private sector firm as a four phase contract. Funding was provided by both the U.S. and German Support Programs to the IAEA for the respective portions of the company responsibilities. After the project start in April 2005, the NGSS project is now nearing its scheduled completion in 2010.

The following paper will outline the project progress and give a comprehensive overview of its achievements up to its current stage. It will begin by outlining the fundamental user requirements that were driving the development and then explain how they were implemented in the first laboratory setups. Next it will present the NGSS prototype units. An overview on the testing schedule and results (as available) will follow. A brief summary of the NGSS fielding and DCM-14 phase out plan will conclude the paper.

THE NGSS USER REQUIREMENTS

The IAEA carefully prepared the user requirements that built the foundation of the development of the NGSS. It conducted not only an internal review to pool the feedback from the end-users (the inspectors), technical support staff, and information management, but also held a user requirements workshop that invited experts from throughout the non-proliferation community, including R&D institutes, support programs, nuclear regulatory authorities, and the private sector. The result was published and opened for comments before the tender for the project was initiated. In late 2004, the project was awarded to a German-U.S. private sector partnership with support from both the German and the U.S. Support Programs. The project was subdivided into four phases, each to be concluded by a design review to accept deliverables and refine the scope of the project for the remaining phases. The following details the four project phases:

- Phase I: Conceptual Design (completed in April, 2006)
- Phase II: Detailed Design (completed in March, 2007
- Phase III: Prototype Development (completed in September, 2008)
- Phase IV: Pre-production qualification testing, start of field testing (on-going)
- Final Acceptance (Category A): Foreseen for July, 2010.

The following key NGSS design goals summarize the user requirements and make the system easy to use, scale, and maintain:

- Short Picture Taking Interval (PTI)
- Support for high resolution and color images
- Support for TCP/IP networking over Ethernet and possible co-existence with current surveillance equipment (backward compatibility)
- Advanced data security (authentication and encryption)
- Integration of the surveillance camera and the security critical components into one tamper indicating assembly
- Modular and fully scalable to allow simpler installation, maintenance, and spare parts logistics
- Low power consumption
- High reliability under harsh environmental conditions
- Commercial-Off-The-Shelf (COTS) and non-proprietary components where possible (extended life cycle management)
- Designed to be easily implemented as Joint-Use-Equipment (JUE).

The IAEA's standard surveillance system DCM-14 was designed in the mid-1990s and is still used in the field today with essentially the same hardware as it was originally fielded. Thus, the lifecycle management for NGSS had to be designed accordingly; the developers were tasked to prepare for 15-20 years of fielding of the NGSS. The component selection of the conceptual design was conducted with this goal in mind.

CONCEPTUAL DESIGN

Initially, the conceptual design focused on the front-end image data generator (the camera) which would drive the format of the generated data and largely define how these data would have to be managed downstream toward the data consolidator (the server) and

then onwards to the review station. To ensure that the critical design goals could be met, selected parts were tested for compliance with environmental standards and radiation tolerance before the conceptual design was presented.

The core of the image data generator, called the Surveillance Core Component (SCC), consists of a board assembly designed around a Digital Signal Processor (DSP) that was selected due to its processing capacity, low power consumption, industry standard ruggedness (e.g., industrial temperature range), and a rich set of integrated, direct memory access-enabled peripherals. This specific DSP was also at the very first stage of its lifecycle, and since it is used in the car industry, it can be expected to be available for an acceptable length of time.

The secondary intelligence of the SCC comes in the form of a micro controller to handle the power management and input/output control functions. The micro controller was selected for its ultra low power consumption, special low-power operating modes, and capability to directly execute programs from its internal flash memory. The micro controller will always be active, even if the DSP is in sleep mode, to monitor tamper detection circuitry and erase cryptographic keys if needed.

For removable data storage, the format of Secure Digital memory cards (SD-Card) was selected. The format has become an acceptable standard with growing storage capacity and acceptable operating parameters.

The firmware of the SCC is based on a especially designed operating system rather than a third party product such as Linux or Windows Embedded. The reasons for this decision lie in the user requirements for NGSS that call for the need of SEU mitigation techniques and a minimum PTI of one second

Originally, the planned approach to generating image data was similar to the DCM-14 which uses JPEG image format with header information to store both visual and state-of-health data. The potentially huge amounts of data generated by the NGSS (a 32 camera system running on a PTI of one second unattended for up to 396 days) prompted the developers to re-think the single image format. Finally, the MPEG-2 video compression standard was selected because of its capability to achieve much higher compression rates by removing temporal redundancy between subsequent pictures.

DETAILED DESIGN

After approval of the conceptual design, the detailed design aimed at providing a working laboratory setup that had as many critical design goals implemented as possible. During the work of Phase II, a major design change in comparison to the DCM-14 took place. Rather than connecting an analog camera to the SCC, the developers proposed to mount a CMOS sensor directly on the circuit board to allow for digital generation of data. This new concept offered a couple of features of high interest to the IAEA.

First, the combination of camera and processing electronics yielded a smaller and more compact design. Second, it allowed integrating all security critical functions into one

tamper indicating module as compared to the DCM-14 where the analog camera and the digital data module (and the cabling in-between) had to be protected as a whole unit. Lastly, it allowed for the use of a high resolution (5 megapixels) CMOS sensor out of which multiple 'camera views' could be clipped at the appropriate image size. Effectively, one single camera with a fisheye lens could record multiple views and potentially replace more than one currently fielded single camera unit.

The data consolidator was designed to replace existing multi-channel surveillance systems (DMOS, SDIS) and systems with separated recording (DSOS). The result was a modular and partly redundant system that provides power, storage and remote access capabilities. The major design features include the following:

- **Redundancy:** Built-in redundancy in critical components shall reduce or, if possible, eliminate single points of failure.
- **Ethernet-based TCP/IP Network Structure:** Ethernet-based TCP/IP provides standardized communication between Data Consolidator sub-modules ,sensors and remote clients.
- **Modularity of Subsystems/Components:** Modularity facilitates maintenance, scalability, and upgradeability. It enables the IAEA to replace subsystems without changing the whole system in the field.
- **Ease of Maintenance:** Routine maintenance such as replacement of batteries, power supplies, and storage devices can be done by an inspector rather than a specially trained technician.

The DC is also designed to store and, in remote monitoring applications, to forward data from all kinds of safeguards sensors, not only surveillance. The networking infrastructure will further allow for cross-triggering of different sensors over the network, combine data from different sensors into a single package, and even performing maintenance, set-up, and troubleshooting up to the sensor level from IAEA HQ, if remote access is agreed upon. The key structure also allows the IAEA to make better use of SSACs or RSACs in supporting their activities. Schemes for joint use of the equipment will be tested during the first field tests.

The NGSS review application will retain the look and feel of the current review tool but allow the inspector to start a review immediately while the processing tools are running in the background, thereby significantly reducing the inspector's waiting time. The review software optimizes the available computer resources. Background tasks are queued and, as resources become available, executed without user intervention. The computer will never be idle until all tasks have been completed. Currently, the computer is idle for a large portion of the time while the inspector manually reviews all images and events. An MPEG stream file viewer was incorporated for the NGSS review. All prototype work completed was implemented in a new version of the General Advanced Review Software (GARS) for in-process testing and to be made available for inspector use and feedback.

NGSS PROTOTYPE

The prototype of the Data Generator introduced a new camera module, a new user interface and a new “Blue Housing”.

The new camera module was named DCM-C5 (Digital Camera Module - 5 Megapixel CMOS Sensor). The DCM-C5 provides a standard C-mount for DC-controlled or manual lenses, an SD-Card slot, a USB service port, an RJ-45 Ethernet jack, an EOSS seal interface, a 16-pin I/O connector, a 7-pin RS-485/Trigger/ext. Power connector (DCM-14 compatible), a user interface connector and connectors for a Li-Ion battery and power input from an isolated AC/DC converter.

Picture 1: DCM-C5 Digital Camera Module

The new user interface module was named DCD (Digital Camera Display). It consists of a TFT monitor for text and video output and a jog dial for user input. The DCD is integrated into the upper lid of the new camera housing. A swivel hinge allows the user to access the DCD from different positions. The DCD is automatically turned on, when the lid is opened and switched off when it is closed.

The rear door of the camera housing provides a holder for a specially designed Li-Ion battery pack. The front door provides access to the camera lens. The prototype camera housing also integrates a class-2 isolated AC power supply.

Picture 2: NGSS Data Generator Prototype

The prototype development phase of the Data Consolidator saw another critical change from the current surveillance approach: each front-end sensor is connected to the data consolidator through a Digital Camera Interface (DCI) to manage communication

functions and to provide power. Each DCI, in turn, is powered by its own dedicated Uninterruptible Power Supply (UPS).

Picture 3: DCI Digital Camera Interface

All other sub-modules in the data consolidator (CPU, Network Attached Storage (NAS), switch, modem, etc.) follow this example, making NGSS a true multi-channel system. An Ethernet network allows the DCI to communicate with the CPU of the system and to offload data onto the system's NAS module. The following figure outlines the NGSS system approach.

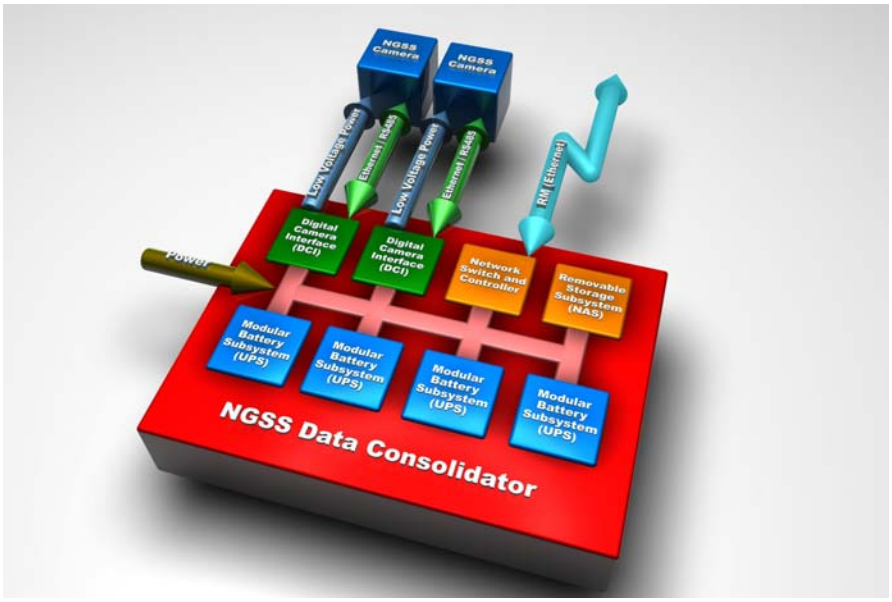

Figure 1: NGSS System Setup

The DCI followed the development of the SCC to some extent by using the same DSP and also the SD-Card format for removable storage. In effect, data are stored three times on an NGSS system: (1) in the camera, (2) then on the DCI, and (3) lastly on the removable NAS device which will consolidate the data of all sensors for inspector removal or remote data transmission. This offers both redundancy and diversity in the data storage.

The processing of the data generated in the SCC includes the following steps: time/date stamping, authentication, and encryption. Furthermore, the data are compressed into the MPEG standard, then forwarded to the DCI and from there passed on to the NAS. The NAS has a removable medium for inspector's use. If the system is operated in remote monitoring mode, the data are transmitted via VPN to the inspectorate. Each module has its independent power supply, removing single points of failure. Even if a submodule fails, data are stored on the camera level for a considerable amount of time. Picture 4 shows the final NGSS Data Consolidator prototype.

Picture 4: Data Consolidator

NGSS TESTING

The IAEA has accepted the prototypes and initiated the zero run production as the major deliverable of Phase IV. Once the prototype units become available, various tests will be conducted that are needed to take the NGSS from development stage to instrumentation authorized for routine in-field use (i.e., Category A equipment). The complete system will undergo environmental testing in accordance with IAEA standards as well as radiation exposure testing for the front-end camera since it will have to withstand unattended operation in high radiation environments. In addition, field testing will be conducted by the IAEA and Euratom to test system performance under real operating conditions.

To understand if the system is compliant with the IAEA requirement of 500 months of mean-time between failure (MTBF), a mixture of MTBF calculation with dedicated software and real in-field data will be applied. The software calculations place the system MTBF at slightly higher than the required mark, but actual in-field data will be needed to measure its actual performance. In order to use the system for inspection use, the IAEA requires a vulnerability assessment to be carried out. Preparations for this are ongoing.

CONCLUSIONS

It is expected that NGSS will be ready for routine deployment in the second half of 2010. NGSS will then start to be the replacement for DCM-14 surveillance systems. The phase in will be gradual and at a pace that IAEA installation teams can support. It is anticipated that DCM-14 systems will remain in the field for up to seven years after the Category A approval of the NGSS. This is due to the fact that some installed systems can only be

accessed at very infrequent intervals (i.e., during reactor shutdown) and also to maximize the return on capital of DCM-14 systems that have been installed only recently.

To facilitate a smoother transition, the NGSS system has been designed so that DCM-14 based cameras can be connected to the NGSS DCI and data consolidator. This allows the inspectorate to change the server and even a selected number of cameras at one time while leaving older units connected for later replacement.

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Methodological Approach for the Assessment of the Performance of Containment and Surveillance Equipment - Trial Application

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Abstract

The inspector's decision process has to be based on safeguards relevant data acquired by using appropriately performing equipment. The ESARDA Working Group on Containment and Surveillance (C/S) is dealing with a performance assessment methodology with regard to C/S equipment. Such a methodology is important for the development of appropriate safeguards approaches and for the reconsideration of existing approaches in view of the implementation of the Additional Protocol and Integrated Safeguards. Furthermore, it can support development of new or improvement of existing C/S equipment. It is assumed that the equipment selection process is facilitated by the ability to determine the performance of C/S equipment. Apart from EC/TREN/Euratom, the users of the assessment methodology would be the International Atomic Energy Agency (IAEA), safeguards designers, plant operators, and instrument developers. The assessment approach foresees the definition of a task profile in terms of the expected functional requirements the equipment has to fulfil in the context of the safeguards approach for a specific facility. Requirement levels are assigned, e.g., specifying whether meeting a requirement would be mandatory or less than mandatory. Then, the performance of competitive C/S equipment is to be checked and, if possible, rated against the defined task profile. On a trial basis the method was applied to a long-term dry storage facility for spent fuel. The paper describes the methodological approach for C/S equipment performance assessment and identifies open questions where further development is still needed for the methodology.

Keywords: containment, surveillance, performance, spent fuel storage

1. Introduction

In the application of International Atomic Energy Agency (IAEA) safeguards at declared facilities, material accountancy is the fundamental measure, with containment and surveillance (C/S) being applied as complementary measures. The basic C/S measures are optical surveillance and sealing. Two roles can be attributed to C/S measures: (1) Facilitating accountancy data verification and (2) ensuring that all material flows pass through key measurement points as declared.

As long as a C/S system does not show any anomaly, re-measuring of the nuclear material at subsequent inspections is not necessary, i.e., the inspection effort can be significantly reduced.

Performance assessment of C/S equipment is important for the development of appropriate safeguards approaches and for the re-assessment of existing approaches in view of implementing the Additional Protocol and Integrated Safeguards. Furthermore, it can support development of new or improvement of existing C/S equipment. The equipment selection process is facilitated by the ability to determine the performance of C/S equipment. Apart from the Euratom Safeguards Directorate, the users of assessment methodologies would be the IAEA, safeguards designers, plant operators, and instrument developers.

Equipment performance aims at the creation of appropriate data in support of the safeguards inspector's assessment and decision process. Performance in terms of detection probability can be determined mathematically for safeguards measures with known measurement uncertainties and statistical sampling procedures. With C/S measures becoming part of a safeguards system, an overall performance figure or detection probability cannot be derived.

The paper describes a non-quantitative performance assessment methodology applied to a facility designed for the long-term storage of spent fuel in casks. It resulted from an account of work performed for and in cooperation with the ESARDA Working Group on Containment and Surveillance [1,2].

2. Methodological Approach of Performance Assessment

The suitability of C/S equipment in the context of a facility-specific safeguards approach in which the equipment is to be deployed, depends on equipment performance. A C/S system will show bad performance for a given task if not compatible with the given conditions. For example, an optical surveillance system monitoring the flow of casks will not render sufficient performance, if there is insufficient lighting inside the facility or if the field of view is too much restricted. A seal will not render sufficient performance, if it is not sufficiently radiation tolerant.

The proposed methodological approach for assessing the performance of C/S equipment includes the following steps:

(1) Acquisition and analysis of the basic technical (and operational) characteristics of the facility under consideration
(2) Assumptions on diversion and misuse scenarios
(3) Assumptions on safeguards approach and definition of safeguards requirements
(4) Compilation and characterisation of candidate C/S equipment
(5) Performance assessment of C/S equipment.

The definition of safeguards requirements results in a facility specific task profile. The performance profile of C/S equipment is defined by the functional specifications and design basis tolerances provided by the equipment manufacturers. In the assessment process, the performance profile of a C/S instrument is checked against the task profile the instrument is intended to fulfil.

Certain requirements may be ‘mandatory’. Candidates of C/S instrumentation that do not meet all of the ‘mandatory’ requirements of the task profile must be, in principle, precluded from the further selection process. The extent to which other task profile requirements are met may be different for each C/S device. Examples are the technical reliability of a device, the effort and time needed for data retrieval and evaluation of results, as well as for maintenance and service of the device.

3. Characterisation of the Facility and Casks

The facility under consideration was designed and licensed for a 40-years interim storage of casks filled with LWR spent fuel as well as casks filled with various types of radioactive waste, in particular, vitrified high-level waste (HLW) resulting from reprocessing of spent fuel assemblies [3-11]. Empty casks may also be stored at the facility. The facility concept foresees no opening of casks and no handling of spent fuel in the event of damage. Therefore, there are no hot cells or any other heavy shielding and remote handling instruments.

The storage capacity is 3.800 tons of heavy metal. There are 420 storage positions where different types of cask are to be placed in an upright position on a base plate. The facility consists of a large hall made of reinforced concrete with a footprint of about 200m by 40m and a height of 20m. The hall is divided into 2 parts (see Figure 1): (I) Reception area and (II) storage area, separated from each other by an approximately 8m high concrete shielding wall with a sliding steel door. The entrance to the facility is located in the reception area; there are two sliding doors, one for incoming and one for outgoing casks on heavy duty trucks as well as for personnel access. In the maintenance room of the reception area, the incoming casks are prepared for long-term storage. Further maintenance work becomes necessary, e.g., in case of a leakage or failure in the pressure monitoring system. An overhead crane is mounted over the entire length of the storage hall and allows the casks to be transported to each of the 420 storage positions arranged in 42 rows of 10 storage positions, respectively.

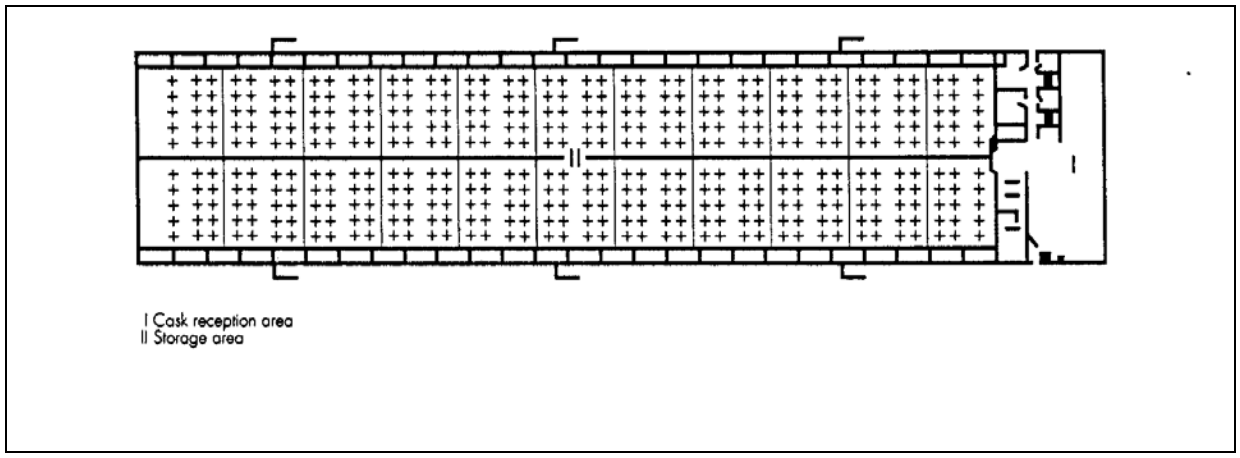

Figure 1: Layout of the Gorleben dry storage facility

Figure 2 shows the length of the storage hall, with newly arrived casks and shock absorbers in the foreground. The arrangement of casks in their storage positions is shown in Figure 3, which also illustrates the dimensions of a cask compared to a human being.

Figure 2: Storage hall with newly arrived casks and shock absorbers in the foreground

Figure 3: Storage hall with storage casks

Transport and Storage Casks

The availability of several licensed cask types enables the storage of various kinds of fuel in different amounts. The design of the different Castor® models consists of a cast iron, thick-walled, cylindrical body with radial cooling fins on the cask surface and a dual lid system comprising a primary and a secondary lid made from stainless steel. Finally, a protective lid is added prior to storage to protect the lid system against mechanical damage. All the three lids are bolted to the cask body as shown in Figure 4. Two trunnions each are placed at the top and bottom ends for handling purposes. A basket with definite loading positions contains the fuel assemblies to be stored. For reducing the neutron dose polyethylene rods are used as moderator integrated into the cask body and secondary lid.

4. Assumptions on Nuclear Material Diversion and Misuse of the Facility

The IAEA makes assumptions on diversion and misuse strategies a state could consider, in order to remove declared nuclear material from or introduce, produce or process undeclared nuclear material in a safeguarded facility. Diversion and misuse scenarios and, accordingly, the safeguards measures to be applied depend on the type(s) of safeguards agreement in force in the state under consideration. If an Additional Protocol (AP) is in force and the state has been evaluated by the Agency with a positive “broader conclusion”, some scenarios may be less relevant than in states without an AP in force.

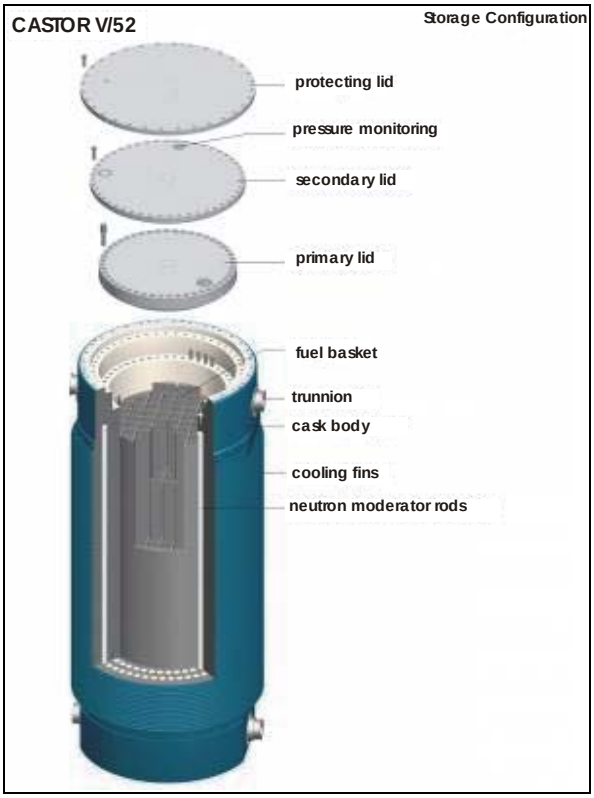

Figure 4: Typical design of a Castor® cask

In the case studied here, no processing of nuclear material takes place, and the removal of a spent fuel cask is the essential diversion strategy. It is assumed that casks are sealed by the inspector in the shipping facility and transported by railway coach and truck to the storage facility. The following diversion scenarios are considered with regard to the reception area:

- Removal of a cask after recording of receipt, e.g., loading it back onto the transport vehicle
- removal of a cask that was retrieved from the storage area for maintenance
- concealing a diversion by replacing the removed cask with an empty cask, a dummy or a HLW-filled cask
- overstate receipts by declaring an empty cask or a HLW cask to be a spent fuel cask.

With regard to the storage area the following diversion scenarios were proposed in the international discussions:

- Removal of a cask via normal access route
- removal of a cask through a hole in the outer wall
- concealing a diversion by replacing the removed cask with an empty cask, a dummy or a HLW filled cask
- lifting of a cask, cutting its bottom and removing all or part of the spent fuel.

5. Safeguards Approach

In the facility under consideration, the spent fuel content of a cask will not be accessible for verification; in IAEA terminology the spent fuel is categorised 'difficult-to-access'. Therefore, it is verified at the shipping facility during its loading into the cask that will be sealed to maintain the continuity-of-knowledge of its content. This leads to the following principles and requirements in the storage facility:

- The dry storage facility is an item facility with the cask being the item.
- Identity and integrity of the cask are to be verified.
- Appropriate C/S measures are to be applied.

Each plausible diversion path is sufficiently covered by C/S. To cope with device failures, i.e., to increase the reliability of the C/S measures, devices may be duplicated or backed up. For instance, a cask is regularly sealed by attaching an electronic seal; the backup seal could be a metallic cap-and-wire seal. In addition, optical surveillance may be applied. The IAEA does not exclude circumvention of a C/S system by a diverter, even if a C/S device shows no anomaly. Therefore, a higher confidence-level is assigned to a multi-layer C/S system in which each plausible diversion path is covered by several C/S devices that are functionally independent and not subject to a common tampering or failure mode. 'Multiple C/S' is normally applied where the verification of nuclear material is difficult to perform, in order to increase confidence in the C/S results and reduce the requirements for periodic re-verification [12].

In the facility studied here, the C/S system is a combination of optical surveillance and sealing. A single camera is deployed for observing the reception area including the maintenance room. As the shielding wall separating the reception area from the storage area does not reach to the roof, it is also possible to monitor the crane in the storage hall, so that all movements of casks and possible diversion and concealment activities can be recorded by the camera system. This includes transfer of casks from the reception area to their final storage positions, transfers within the storage area from one to another position, and transfers back to the reception area. The camera would also record a possible cask removal and/or replacement with a substitute as well as the unrealistic case of cask opening to remove spent fuel.

Identity and integrity of each cask are verified upon arrival at the storage facility by verifying the safeguards seal on the secondary lid. As a backup measure for the subsequent storage period, a metallic seal is commonly installed by an inspector in exchange of the seal used during transport from the shipping facility. Later on, this seal will be covered by the protective lid and only be accessible upon removal of the lid. The main seal in the storage facility is, however, attached to the protective lid ensuring the continuity-of-knowledge in terms of nuclear material content, identity of the cask, and integrity of its lid system. In addition, a seal may be applied to a group of casks, in order to ensure the immobilisation of the casks, thus serving as a back-up measure in case of a camera failure. It will also reduce the irradiation dose associated with inspections.

6. Task Profiles and Performance Profiles

The surveillance camera is assumed to be overlooking the whole length of the building (200m), and to monitor the operations carried out on the cask after entering into the reception area, such as detachment of 'transport seal', installation of protective lid, and attachment of 'storage seal'. The facility operator provides power to the camera; a battery back-up is available to bridge mains power outages.

As the facility operator will enter into the storage hall only if necessary for operational reasons, i.e., in order to emplace a new cask or carry out maintenance or routine check-ups, it is desirable to reduce the lighting level without compromising surveillance. Data capture and storage capacity of the camera system are required to cover at least 3 months of unattended operation.

In the assessment it is intended to compare the performance levels achieved by different types of C/S instrumentation for the same application. The following ratings are applied:

no compliance	not relevant	Satisfactory	good
-1	±0	+0.5	+1

If any 'mandatory' requirement has a performance rating of -1, the device will have to be excluded from application. If, however, there is no alternative C/S instrument available, then the feasibility of an appropriate maintenance and replacement strategy will have to be studied.

Optical Surveillance

The optical surveillance systems approved by the IAEA for routine safeguards use are based on the DCM14-technology [13]. In our study, the single-camera system ALIS (All-In-one-System) [14] is the only available candidate.

Task Profile / Requirements	Requirement Level	C/S Performance Rating
Ability to focus on short distances of a few meters up to long distances of about 200 meters (minimum size of object to be identified at 200m distance, e.g., 6m×2.5m)	Mandatory	+1
Data capture and storage capacity for at least 3 months with the desired picture taking frequency	Mandatory	+1
Ability to operate under reduced lighting conditions	High	+1
Operating on mains power supplied by the operator, with battery backup	Mandatory	+1
Tamper indication	Very high	+1
Availability of powerful support to facilitate the picture evaluation	Very high	+1
System reliability (IAEA target: 150 months MTBF)	High	+1
Ease of use (usability review passed)	High	+1
Interface with electronic seals	High	+1
External triggering	High	+1
Embedded time stamp	Mandatory	+1
Remote monitoring capability	High	-1
Status of Health (SoH) information	Mandatory	+1
End of life	AD 2018	+1

Table 1: Performance Assessment of Optical Surveillance

In view of the dimensions of the building, two ALIS cameras “looking at each other” might be a better solution than one, in order to cope with before-the-lens tampering. Whereas the first camera, installed in the reception area, could possibly overlook human beings that are determined to hide in the storage hall, a second camera would spot a human being, even in remote corners and within cask shadows.

Table 1 shows that ALIS has no remote monitoring capability. Such a capability would enable the retrieval of state-of-health data from the camera, but also of image data for safeguards. Any optical surveillance failure would trigger immediate remedial action on the part of the inspectorate, while there will still be a functioning C/S system in place, as the casks will be sealed. For remote monitoring, a DMOS (Digital Multi-camera Optical Surveillance System with up to 16 cameras) or SDIS (Server-based Digital Surveillance System with up to 6 cameras) could be implemented instead of ALIS. For instance, the use of a two-cameras SDIS would increase the review effort, but it may also increase the transparency of plant operations.

Sealing

In the present paper, only the sealing of the protective lid is discussed. The following seals have been selected for evaluation: Metallic cap-and-wire seal [15]; COBRA seal [16]; VACOSS seal [17].

The seal will be accessible at any time. According to the safeguards approach, it provides for verification of cask identity and integrity. It has to be properly applied by threading the sealing wire through at least two bolts in such a way that the bolts cannot be unscrewed without damaging the sealing wire.

The effort required to verify the seal can vary considerably depending on the seal type. If a cap-and-wire seal is used, the integrity of the whole length of the mechanical wire has to be checked visually, i.e., the inspector has to climb to the top of the cask, i.e., about 6 meters above ground. There must be enough space to place and operate a lifting platform (or a ladder) between the cask rows. Furthermore, the length of the stay in close vicinity of the casks will increase rapidly with the number of casks to be verified. This implies that inspectors and operator’s staff are exposed to high radiation doses.

The same considerations hold true for the COBRA seal. Its sealing wire consists of a fibre optic cable. The manufacturer of the COBRA seal recommends that the whole sealing wire should be checked for its integrity, as there are tools available to cut and repair it. This would not be detectable by using the seal reader. It should be noted, however, that repairing the sealing wire would be time-consuming and, thus, imply significant exposure to radiation.

The electronic VACOSS seal, too, has a fibre optic cable as sealing wire. As it is monitored by the seal electronics, it is more resistant to undetected cutting and repair. Technological advances may not exclude that cutting and repair could be a viable option for concealment. In contrast to the cap-and-wire and COBRA seals, VACOSS offers the great advantage of remote interrogation from outside the storage area with a serial cable interconnecting up to 40 seals. With this approach significant radiation exposure can be avoided by inspectors and operator's staff. One could even think of a VACOSS sealing approach which works similar to the operator's pressure and leakage monitoring system, i.e., an unattended system recording alarms in the event of undesirable status changes like low seal battery and opening of the sealing wire. While the VACOSS seal is reaching its end of life, the IAEA is already replacing VACOSS seals with EOSS seals [18].

Task Profile / Requirements	Requirement Level	C/S Performance Rating		
		Cap-and-Wire	COBRA	VACOSS
Seal wire diameter less than 7 mm	Mandatory	+1	+1	+1
Wire integrity checked by seal device (no human visual inspection required)	Very high to medium	-1	-1	-1
Maximum wire length at least 10 meters	Very high to medium	+1	+1	+1
Wire can be fixed in bolt drilling so that bolts cannot be unscrewed even with long sealing wire	Mandatory	+1	+1	+1
Operating temperature between 0°C and 80°C (max. temperature of cask surface)	Mandatory	+1	+0.5	+0.5
Unattended operation for up to 1 year	Mandatory	+1	+1	+1
In situ verifiable	Very high to medium	-1	+1	+1
Capable to function in high-level radiation field (wire and seal device) [consider this requirement together with replacement frequency]	High	+1	-1	-1
Remote interrogation of seal possible	Very high to medium	-1	-1	+1
Chaining of seals for remote interrogation possible	Very high to medium	-1	-1	+1
Low health impacts (radiation exposure) for seal evaluation for inspector and operator * [this requirement is related to the previous ones, i.e., remote interrogation]	Very high	-1	-1	+1
Low interference with plant operation in terms of required manpower and equipment support from operator [this requirement is related to the previous ones, i.e., remote interrogation]	High	-1	-1	+1
Low effort for seal and wire maintenance in field	High	+1	+1	+1

Task Profile / Requirements	Requirement Level	C/S Performance Rating		
		Cap-and-Wire	COBRA	VACOSS
Low effort for seal and wire maintenance at HQ	High	0	0	0
Low effort for seal evaluation in field	High	-1	+0.5	+1
Low effort for seal evaluation at HQ	High	-1	+1	+1
System reliability **	High	+1	+1	+1
Ease of use	High	-1	+0.5	+0.5
Low false alarm probability	Very high	+0.5	+1	+0.5
Embedded time stamp	Very high	-1	-1	+1
Status of Health (SoH) information	Very high	-1	-1	+1
End of life **	AD 2030	+1	+1	-1
Cost	investment operation	+1 -1	0 0	-1 +1
Supplier(s) and Procurement	***	+1	+1	+1
	Sums	**-0.5**	**+4.5**	**+13**

Table 2: Performance Assessment of Seal on Protective Lid

* Depending on the number of casks stored in the hall, the duration of the inspector's access may be limited by the radiation dose.

** Maintenance and replacement strategy to be applied.

*** There must be at least one manufacturer for each type of C/S device to assure the supply. For customised equipment, the intellectual property rights must be vested either with the inspectorate or with a Member State Support Programme to the IAEA.

The performance assessment renders, **in principle, all of the seals suitable for this application**. If deemed necessary, it would be possible to apply two seals with different failure modes. For reason of comparison, the ratings have been summed up in the last row of Table 2. The results suggest that the electronic VACOSS seal, showing the highest rating, would be the most appropriate seal for this application. The advantages of this seal are mainly due to its remote interrogation capability and the consequences thereof, i.e., evasion of radiation risk.

7. Summary and Conclusions

The simplified example of an interim dry storage facility for spent nuclear fuel and high-level waste has revealed that there are many parameters that influence the performance evaluation of C/S instrumentation. The number of casks stored in the facility plays a major role. If there are only very few casks to be verified, the application of metallic cap-and-wire seals for casks may be an adequate and robust solution. But, the capacity of the studied facility is 420 casks.

The definition of requirement levels was arbitrary, and different assessors may come up with different level assignments. Besides problems of defining the level of a requirement and rating the level of fulfilment by an individual C/S instrument, there are also methodological problems still to be solved, e.g., the problem of comparing and unifying the different nature of factors. How can effects of equipment costs, health impacts on persons due to radiation exposure and levels of reliability be balanced with each other?

It would also be interesting to analyse retrospectively, how verification activities were carried out in practice in comparable facilities and to analyse the practical differences for different facilities. With regard to safeguards assurance, the IAEA does not seem to rely totally on C/S application. In their evaluation criteria for safeguards measures under INFCIRC/153-type agreements the IAEA requires for material under conclusive single C/S and even dual C/S a certain degree of re-verification. Given

identical C/S systems, the strategic value of the material safeguarded by this system seems to play a role in assigning safeguards assurance to the results.

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Towards Remote Safeguards Inspections

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Abstract:

Remote safeguards inspections (RSIs) can be defined as inspection activities based on the analysis of data acquired from the field without the physical presence of an inspector in the field. The purpose of RSIs is to optimize the IAEA's inspection effort, while maintaining effective safeguards implementation. On-site inspections will remain essential, but their contribution to overall safeguards approaches could be further optimized by the use of RSIs, allowing for a shift of inspector resources from routine activities to non-routine activities (e.g. complementary access and searching for indications of undeclared nuclear material and activities).

Principally, RSIs are implemented utilizing unattended data collection systems for containment and surveillance (C/S) and radiation (NDA) sensors. The data acquired is transmitted via secure remote transmission technologies to remote centres (IAEA Headquarters in Vienna or IAEA Regional field offices), where it is analyzed by expert teams of inspectors who are able to draw safeguards conclusions and provide feedback to the field (including instructions for follow-up action, if necessary). The second possible means of implementing RSIs is through making full use of State and regional systems of accounting for and control of nuclear material (SSACs/RSACs) whereby State inspectors and/or facility operators operate attended inspection systems with remote and secure IAEA oversight.

The remote transmission of operators' nuclear material declarations on a timely basis while at the same time meeting the highest IAEA standards and requirements is key to the successful implementation of RSIs. Remote inspection activities are especially important in the case of modern facilities where processes are increasingly automated and where access to nuclear material is virtually impossible. In such cases, the verification role of the inspector should be fulfilled through remote inspection activities to the greatest possible extent.

The paper describes elements of RSIs which are presently implemented in various facilities such as reprocessing plants, research reactors and plutonium fabrication plants. With the expected availability of new detection technologies, advanced C/S equipment and enhanced information technology (IT) capabilities, RSIs will play an increasingly important and versatile role in future safeguards implementation.

Keywords: IAEA inspections, remote monitoring, integrated safeguards, unattended monitoring systems, SSAC.

Introduction

Remote safeguards inspections (RSIs) can be defined as inspection activities based on the analysis of data acquired in the field without the physical presence of an inspector. The purpose of RSIs is to optimize the IAEA's inspection effort and to reduce the presence of inspectors in the field while at the same time maintaining credible safeguards implementation. On-site inspections and visits by inspectors will remain essential, but their contribution to an overall safeguards approach can be optimized through the use of RSIs. The RSI makes use of unattended data collection systems for containment and surveillance (C/S) and radiation (NDA) sensors. The acquired data is transmitted using secure remote transmission technologies to HQs (in Vienna or in Regional offices) for analysis by expert teams for the purpose of deriving safeguards conclusions and providing feedback to the field or instructions for follow-up action, where necessary. Where possible, RSIs will make full use of State and Regional systems of accounting for and control of nuclear material (SSAC/RSACs) allowing State

inspectors and/or facility operators to operate attended inspection systems under remotely controlled IAEA mechanisms. The timely and remote transmission of operators' nuclear material declarations meeting highest IAEA quality standards is a key success factor in the application of RSIs.

Elements of RSIs, based on unattended monitoring systems (UMS) and C/S systems operated in remote monitoring mode, are already partly implemented in various facilities worldwide (such as reprocessing plants, research reactors and plutonium fabrication plants). With the availability of new detection technologies, advanced containment and surveillance equipment and enhanced information technology (IT) capabilities, RSIs will play an increasingly important and versatile role in future safeguards implementation.

Remote inspection activities are especially important for facilities where processes are highly automated and access to nuclear material is virtually impossible. In such cases, the verification role of the inspector should be taken over by remote inspection activities to the maximum possible extent.

Distribution of Inspection Effort

In 2008, the Agency applied safeguards to a total of 1131 facilities including 445 locations outside facilities (LOFs) in ~80 countries worldwide. The variety of facilities and LOFs is presented in Fig.1.

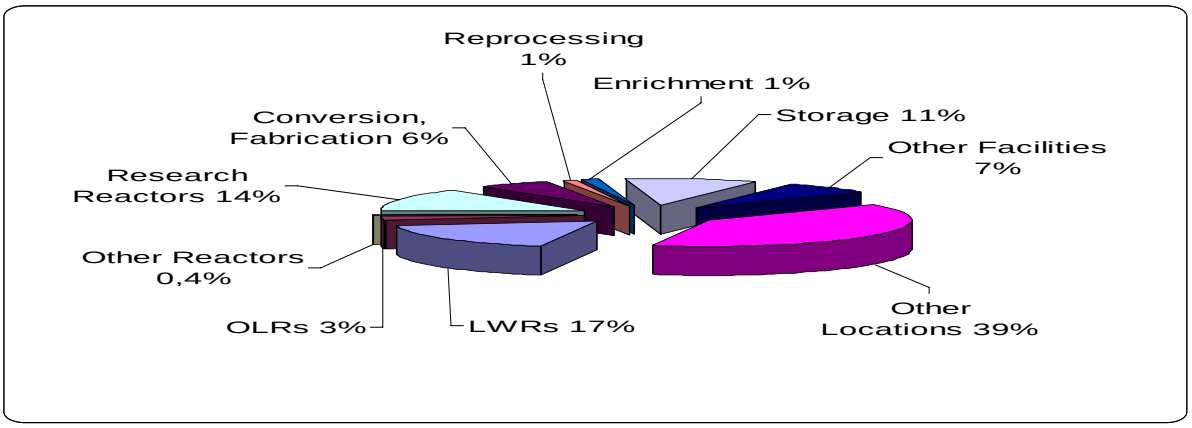

Figure 1: Facilities and LOFs under Safeguards in 2008

The inspectors carried out 8221 person days of inspection (PDIs) in the field, with 90% of this effort being spent in only 20 countries. Fig.2 illustrates the distribution of inspection effort per facility type in 2008. Bulk handling facilities (reprocessing, conversion, fuel fabrication and enrichment) account for about 8% of the total facility population but the respective inspection effort amounted to ~40% of total inspection effort.

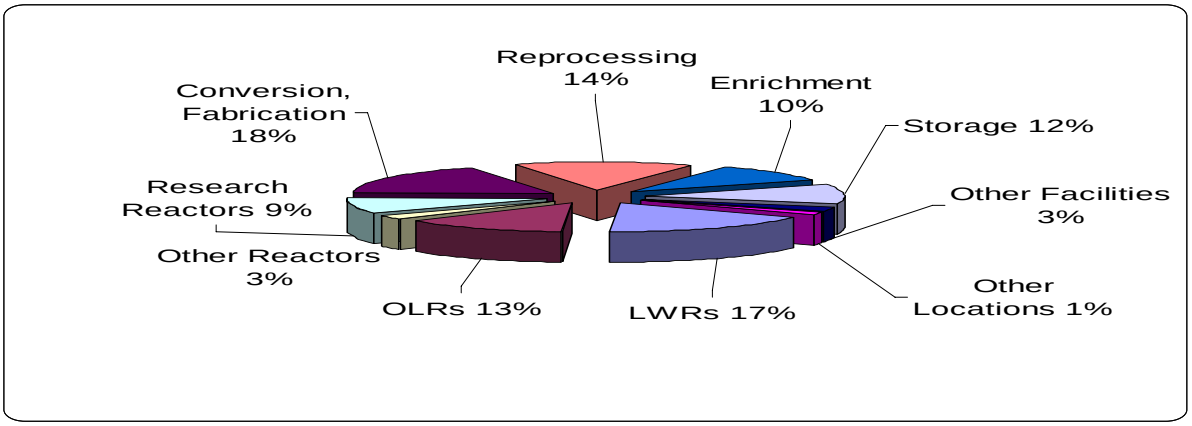

Figure 2: Inspection Effort (PDI) per Facility Type in 2008

The impact of replacing inspection effort on a facility type basis by RSIs is therefore most significant. RSIs could substantially reduce the inspection effort for on-load reactors (mainly Candu) which require about five times the inspection effort of an off-load reactor. Resultant savings in inspection effort could help to cope with an expected increase in nuclear power generation activities and safeguarded materials arising from the forecast 'nuclear renaissance', the anticipated increase in safeguards verification activities in India as well as in some nuclear weapon States, thereby reducing reliance upon a proportional increase in budgetary resources. Expanded usage of MOX fuel at off-load reactors and the foreseeable increase in transfer of spent fuel to dry storages will also add significantly to the IAEA's 'inspection burden' — a shift of routine activities in the field towards remote inspection activities could free up in-field inspectors to carry out non routine activities such as complementary access (CA) and searching for undeclared nuclear material and activities.

Remote Safeguards Inspections and Integrated Safeguards (IS)

Integrated safeguards has become the new standard in IAEA safeguards. Provided that the IAEA is able to draw and maintain a "broader conclusion" regarding the completeness and correctness of a State's peaceful nuclear programme, inspection resources can be optimized and on site inspection activities reduced. Inspection results in an IS scheme are being evaluated within the context of all other available information (e.g. open source, satellite imagery, complementary access activities etc.). Under IS inspection regimes, on-site inspection effort is significantly reduced by extending the timeliness goal for spent fuel verification from the present three-month to a twelve-month interval, and through applying lower detection probability and defect testing levels. Inspection frequency could be lowered further by applying unannounced inspections. Where unannounced inspections cannot be implemented, short notice random inspections (SNRIs) may be implemented supported by surveillance in 'overwrite mode' or images triggered on demand.

RSI is another important technical tool under IS approaches to achieve the objective of delivering timely safeguards data with a reduced requirement for the on-site presence of IAEA inspectors. Reduced IAEA in-field inspection activities achieve a commensurate lowering of effort and intrusiveness for facility operators and SSACs (e.g. less frequent requirements to provide escorts and reduced operational interruptions). A cost-benefit analysis on a case by case basis, considering all boundary conditions, State specific factors, capital costs and human resource requirements for implementation and maintenance, is needed before an optimized integrated safeguards approach can be applied.

The role of State and Regional Systems of Accounting for and Control of nuclear material (SSAC, RSAC) and Facility Operators under Remote Safeguards Inspections

Even with the most sophisticated verification system, the IAEA cannot fulfil its mission without cooperation with the SSAC/RSAC. Article 7 of the comprehensive safeguards agreement (CSA) [1] stipulates that the Agency, in its verification activities, shall take due account of the technical effectiveness of the State's system. An effective, technically competent and independent SSAC/RSAC is a valuable partner during joint inspections. In the case of remote safeguards inspections, an SSAC/RSAC might even provide its own inspection resources to carry out activities under the direction of the IAEA inspectorate, provided that the necessary independence is maintained. Examples of such enhanced cooperation are already implemented whereby the IAEA controls an inspection schedule, which is jointly implemented. The IAEA inspectorate has the right to participate in selected individual inspections but may decide on an unannounced, short notice and unpredictable basis to forego its participation in any single joint inspection. This concept might be further developed by directing and observing the national inspectorate via remote communication means to carry out actual measurements on the IAEA's behalf using additional devices that ensure authentication of the verification activity (e.g. time and location stamping). An SSAC/RSAC could also initiate the remote transmission of inspection data collected from unattended systems upon IAEA demand, or could simply collect verification data from an unattended system and provide it to the IAEA inspectorate. Appropriate system configuration and other measures should ensure that the data is transmitted without compromising data authenticity and data sharing protocols.

Facility operators could also perform selected activities for RSIs, for instance, in attaching electronic seals on nuclear material containers under surveillance (such as reactor vessels, spent fuel casks or UF6 cylinders). The attached seals would be subject to verification by the inspector at a later stage or location. Of course, while inspection efforts and costs under such approaches would be reduced for the IAEA, more of a burden would be placed on the operators and their State authorities.

Equipment for Remote Safeguards Inspections

A variety of unattended monitoring systems have been developed and installed at numerous facilities worldwide in order to verify nuclear material movements and inventories, and to maintain continuity of knowledge (CoK) of nuclear material and activities. Unattended systems usually monitor 100% of items and process flow. These systems have the major shortcoming of being predictable and therefore must be supplemented by additional verification measures. Many NDA systems work in concert with C/S. These are designed to be highly reliable, providing authenticated safeguards data on the status of nuclear material flows and inventories. Material balance areas (MBA) and key measurement points (KMP) are carefully selected to match the results of unattended monitoring and surveillance with an operator's declaration.

UMS applications can range from tracking items such as spent fuel bundles or assemblies to performing quantitative assays (e.g. determining Pu content of canisters of mixed oxide fuel). The advanced power monitor is another example of an unattended system which is used in research reactors to accurately determine plutonium production capabilities.

The deployment of UMSs is especially important in areas that are difficult for inspectors to access, such as automated MOX and reprocessing plants, where these systems are often in continuous operation. While C/S measures secure the time horizon of monitoring activities in a specific location, NDA accounts for the specifics of the various nuclear materials under safeguards. For example, all entry points to a material balance area are monitored continuously by surveillance systems, and sensors detect the presence, type, and quantity of nuclear materials by measuring neutrons, gamma rays or weights.

Remote monitoring is critical in enabling the inspectorate to carry out remote inspection activities. In remote monitoring applications, UMS and C/S systems are able to transmit safeguards data in a secure manner from an inspected facility to a data centre located virtually anywhere in the world. The remote transmission of verification data to IAEA Headquarters allows for its speedy evaluation by an expert group and central archiving, thus enabling the inspectorate to perform better follow-up and post analysis. The timing of the remote data transmission can be either in real time or upon demand. Given the eight hour time difference between the IAEA's Vienna Headquarters and, for example, its Regional field offices or facilities in Japan or Canada, the expert group at HQ is able to evaluate the transmitted data while the in-field inspectors are off duty. The current suite of data evaluation tools includes: Radiation Review (graphical plot of radiation signals against time), Digital Video Review (GARS — displays video images against time/event), Neutron Coincidence Counting (INCC —quantitative analysis of neutron data), Plutonium Isotopic Review (MGA — quantitative analysis of isotopic composition) and NRTA. It is expected that enhanced IT could in the future provide an integrated expert tool for the automated review/evaluation of remotely acquired data.

The operator can also provide his declarations on nuclear materials and activities using mailbox systems which could also serve as a basis for performing book audit activities. Face to face communication with the facility operator could be readily established where necessary using video conferencing services. The sharing of remote inspection data with State authorities requires specific conditions to be met but can be readily implemented from a technical standpoint. Furthermore, remote monitoring provides the capability of assessing the operational status of verification instrumentation and, in many cases, malfunctions can be repaired remotely without the need for technicians to visit a facility.

Table 1 shows the present status of systems connected remotely to IAEA Headquarters in Vienna. Most of the systems are connected via the internet using secure VPN tunnels and the transmission of large quantities of data can be performed at very low cost. There is a significant potential for an increasing number of systems to be remotely connected. However, the implementation of remote

technologies still meets with either a lukewarm response or no support in various countries, limiting its possible application.

Instrument	Type	Fielded	RM (data)	RM (SoH)	% (SoH)
SDIS	Surveillance	115	55	82	71
DMOS		29	23	24	83
ALIS		205	0	0	0
DSOS		105	0	0	0
BDIS/QDIS		33	0	0	0
Subtotal:		487	78	106	22
VIFM(B/C)	Unattended NDA	46	32	32	70
UFFM		24	19	24	100
ATPM/GRPM		10	1	1	10
PCAS		4	2	2	50
RRP systems		14	14	14	100
Others		34	6	6	18
Subtotal:		132	74	79	60
VACOSS	Seal	911	56	56	6
EOSS		52	1	1	2

Table 1 : Remote transmission status of UMS (end of 2008)

New measurement technologies can be incorporated into unattended systems to perform real-time process monitoring at declared facilities with very high reliability. Standardized and integrated platforms for NDA and C/S data collection could also be designed in such a way to allow installation, service and preventive maintenance by third parties without the presence of IAEA personnel. The next generation of surveillance systems (NGSS) already has a secure modular design, whereby security sensitive modules are intrinsically tamper indicating. This allows handling of the modules without the need for the physical presence of IAEA inspectors.

Smart surveillance systems supported by object recognition for safeguards relevant items could complement safeguards monitoring activities. Radio frequency identification tags (RFID) for identification and tracking of items are widely used in industry and could potentially be used for RSIs, provided that all associated safeguards vulnerabilities are mitigated. A network with various, multiple sensors could produce verification/monitoring matrices for various processes which are more difficult to defeat than stand-alone systems. A new generation of attended systems may play a role in RSI in the framework of IAEA cooperation with SSAC/RSACs, provided that these attended instruments could be modified to recognize tagged items and to produce authenticated measurement reports.

Remote Safeguards Inspections at Reactor facilities

Reactor facilities are item-accounting facilities and the associated safeguards measures are targeted primarily at providing CoK of the fresh and spent fuel (SF) held in the core and spent fuel ponds. Containment and surveillance measures over spent fuel ponds, hatches and reactor cores are the main verification tools used to perform this task. C/S systems are intrinsic unattended systems and, for a number of countries, are operated in a remote monitoring mode (e.g. all LWRs in the Republic of Korea). Remaining inspection activities such as book audits, fresh fuel verifications, design information verifications (DIVs) and spent fuel (SF) transfer verification are normally performed during on-site inspection.

SF transfer verification requires intensive efforts to maintain CoK during loading and shipment. Substantial increases in cask loading operations for SF are anticipated in the future, and more and more SF transfer verifications are likely to be performed by unattended NDA systems in combination with C/S measures. As for facility-specific safeguards approaches, an approach based on randomized unannounced inspections and remote monitoring to verify spent fuel transfer from an on-load reactor (OLR) to dry storage facilities is implemented at two OLR sites, resulting in considerable savings in

inspection effort. Unattended radiation monitoring systems placed in transfer routes and other equipment serve to distinguish fresh, spent and dummy fuel during its movement from storage to the reactor core, as well as the discharge of spent fuel into storage pools. For example, such systems have been installed in Candu reactors in Canada, Romania and Republic of Korea. These systems are also applied (in different configurations) at Fast Breeder reactors (FBRs) in Japan (MONJU, JOYO). It is extremely difficult and resource intensive to recover from failures of UMS used for FBRs, which stresses the absolute need for the highest reliability levels and redundancy value. Presently, efforts are underway to route verification data from various stand-alone monitoring systems, including C/S systems, to a central data collection platform, which could further distribute this data to off site locations in remote monitoring mode, thereby providing the basis of a remote inspection scheme.

The future use of MOX fuel in LWRs is another area expected to increase the future inspection efforts. Remote monitoring of fresh MOX fuel assemblies during storage and loading of MOX fuels will potentially be a major tool to optimize this associated inspection effort.

Monitoring of operational parameters (e.g. power monitoring or boron concentration at LWRs) may provide powerful additional safeguards strengthening measures.

Remote Safeguards Inspections at Enrichment Plants

Safeguards activities at enrichment plants aim to verify the material balances for feed, product and tails, to confirm the absence of undeclared LEU production and that no highly enriched uranium is being produced. Material balance verification is mainly achieved by attended systems using item tracking and weighing of UF6 cylinders in process and storage areas with subsequent DA sampling and NDA being implemented on a random basis. The facility operator has many instruments distributed across an entire plant, for example load cell systems. Investigations are underway to determine the optimal conditions and benefits which might be associated with the joint use of operator weighing systems (such as feed and take off station load cells and accountancy scales). This economical option for the IAEA implies the organization's early involvement in system design and innovative measures for data authentication being put in place. Recently, a laser based unattended monitoring system has been demonstrated which has the ability to uniquely identify UF6 cylinders being transferred into the process. Together with surveillance measures, such data can be provided remotely to off-site locations. Expansion of these safeguards measures through the use of an unattended NDA system for UF6 cylinders could further decrease reliance on in-situ inspection activities.

An affordable verification solution to perform the monitoring of cascade areas is still to be developed. Advanced enrichment monitoring on the piping is under consideration in combination with load cell monitoring. The existing security concerns of operators and State authorities regarding the remote transmission of process data beyond their obligatory nuclear material accountancy declarations also need to be addressed. In practice, this means that load cell data and accurate assay data on UF6 pipes needs to be processed on site utilizing mutually agreed methodologies and software to ensure that only safeguards relevant data is remotely transmitted to the IAEA.

A qualitative continuous enrichment monitor (CEMO) has been operated for some considerable time in an enrichment facility to remotely deliver to IAEA Headquarters "Yes/No" information regarding the presence of HEU. CEMO technology needs to be upgraded to ensure affordable and full scale deployment at large Gas Centrifuge Enrichment Plants.

Remote Safeguards Inspections at Reprocessing Plants

The application of safeguards at reprocessing plants is a most complex and challenging task for the IAEA and requires major significant inspection efforts. There is therefore a great potential to save inspection resources through applying remote inspection measures. The Tokai Reprocessing Plant (TRP) has developed several instruments for remote inspection use such as FTPV (Spent Fuel Transfer Pool Video System), SMMS (Solution Measurement and Monitoring System), HMMS (Hull Measurement and Monitoring System) and VWCC (Vitrification Wastes Coincidence Counter) [2]. Most of these systems are integrated into the process line and, coupled with appropriate C/S measures, are the basis for process monitoring within these facilities. The remote data transmission

facilitated by these systems has reduced inspection efforts as well as intrusiveness for the plant operator.

For example, the Japanese Rokkasho reprocessing plant (RRP) uses extensively unattended measurement systems complemented by C/S systems to monitor the main process flows [3]. As a matter of fact, the present SG approach employs 14 different unattended systems with about 70 surveillance cameras in place. All of these unattended systems, some of them using neutron detectors and high resolution gamma systems, collect and transmit verification data to a central server in real-time and allow for the independent verification of operator declarations. By employing remote data transmission, a significant part of the data could be reviewed/evaluated off-site, allowing for the redirection or reduction of the workload of in-situ inspectors towards other verification activities.

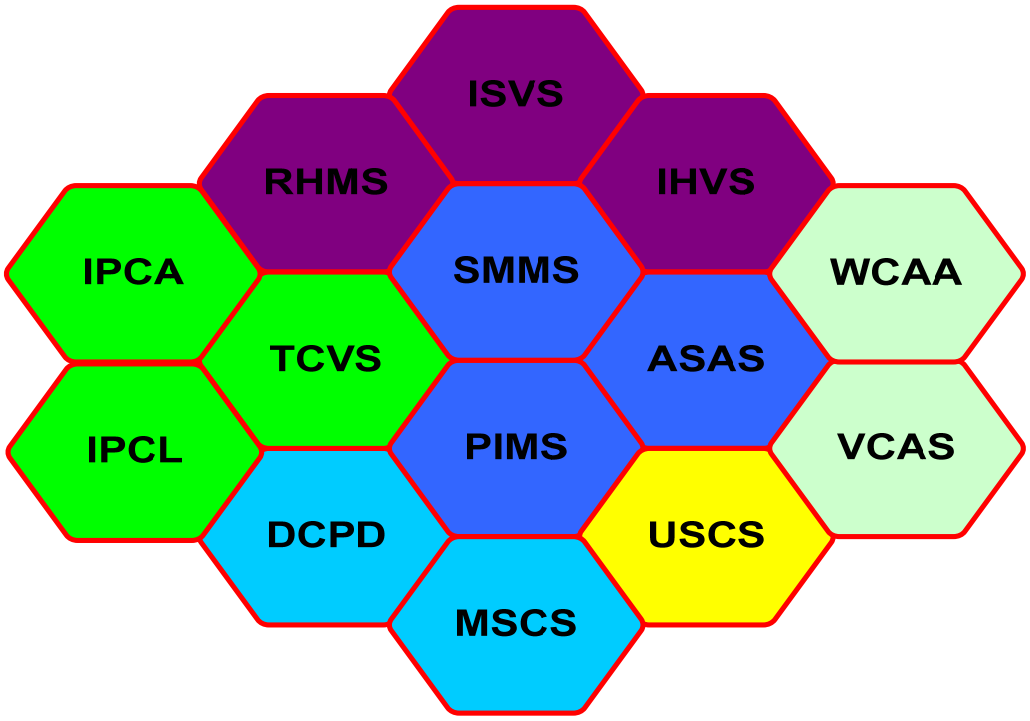

Figure 3: RRP unattended monitoring systems

Remote Safeguards Inspections at Plutonium Fuel Fabrication Plants

Another inspection intensive area for the IAEA is plutonium fuel fabrication including MOX conversion. The Plutonium Fuel Processing Facility (PFPF) in Japan uses advanced UMS systems with neutron coincidence counter detectors to quantitatively monitor nuclear materials during its process as feed material arrives, is introduced to the process and leaves the facility [4]. Radiation detectors track the movement of radioactive materials and video equipment provides for CoK. Each of the UMS components transmits its collected data over a network to a “master multi-instrument collector computer” that monitors the performance and communication status of each component. The major systems in use are the Plutonium Canister Assay System (PCAS), Material Accountancy Glovebox Systems (MAGB) and the Fuel Assembly Assay System (FAAS). UMS and C/S data acquired can either be retrieved on site or sent using remote transmission to a regional IAEA inspection office in Tokyo or to IAEA Headquarters in Vienna. The data from different sensors is automatically compared for internal consistency and against electronically transmitted operator declarations. For example, the movement of a plutonium canister as declared by an operator is identified on video, measured by neutron detectors and followed by motion sensors. The associated software application performs pattern recognition through comparing all relevant signals and alerts the inspectorate in the case of deviations, which can then investigate the flagged events.

Efforts are underway at one site for all plutonium handling facilities to integrate their UMS and C/S systems into a centralized system to reduce and optimise inspector presence at the individual facilities.

Presently, there is a new MOX plant under construction at the RRP site. The safeguards approach for this plant will rely extensively on unattended inspection activities utilising state-of-the-art monitoring systems with subsequent remote data transmission capabilities.

Remote Safeguards Inspections at LEU Fuel Fabrication and Conversion Plants

LEU fuel fabrication and conversion plants are usually covered by short notice random inspections performed during normal working hours. Most of the flow verification (mass balance verification) is carried out using attended instruments. Remote inspection activities are presently focussed on mailbox evaluations. Consideration could be given to automated receipt verification of feed powders and for the final assemblies. Continuous UMS at key points in the process (e.g. sintering furnace) will be a step forward towards ensuring that the plant is operated as declared.

Remote Safeguards Inspections at Storage Facilities

The IAEA has successfully implemented at a plutonium storage facility a remote monitoring system based on surveillance coupled with radiofrequency seals. The nuclear material is continuously under dual C/S and on site inspection efforts are mainly devoted to maintenance activities and placing additional materials under seal. One new technology for the future is the secure sensor platform, which is geared towards enhancing the capabilities of fiber optic loop seals, as well as the development of a tiny gamma spectrometer system for radiation monitoring applications.

The use of unattended monitors to perform the verification of spent fuel receipts within dry storages provides additional savings of Agency inspection effort in the field.

Remote Safeguards Inspections at Final Repositories

Remote inspection activities seem the only viable solution for safeguarding final repositories. The present methodology using laser scanning, surveying, seismic monitoring and satellite imagery has already been demonstrated. The safeguards elements associated with DIV and area monitoring may contain new types of geodetic, geophysical and remote sensor techniques that have to be adjusted and applied according to the particular geological site and facility type.

Conclusions

Remote safeguards inspections will play an increasingly important role in IAEA safeguards as part of efforts to maintain or increase effectiveness and efficiency without increasing resources allocated to inspectors or overall verification costs. This approach requires the expanded deployment of UMS and surveillance systems for the remote collection and transmission of verification data. SSAC/RSACs could contribute significantly to the implementation of RSIs. The evaluation of safeguards data from RSIs requires powerful IT tools to enable the inspectorate to cope with the increased amount of data generated there from and to translate the savings of on-site inspections into real savings in safeguards inspections. The existing security concerns of operators and State authorities regarding the use of remote monitoring need to be addressed in order to fully exploit the great potential of remote safeguards inspections.

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Implementation of JRC Candu Sealing Systems in Cernavoda (Romania) and new developments.

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Abstract:

This paper presents the current implementation status of the JCSS (JRC Candu Sealing System) in Cernavoda (Romania): seals, reading system, seals database. We are discussing also the lessons learned during two years of experimentation together with Safeguards inspectors. A description of the upgraded system that is going to be deployed at Cernavoda-2 is presented.
On-going developments of various JRC Ultrasonic Sealing Systems for both underwater and dry spent fuel storages applications, in particular for dry storages using cask with concrete biological shielding cover are also presented.

Keywords*: Spent fuel storage, ultrasonic sealing bolts, underwater seals, CANDU design, Cernavoda (Romania)*

1 Introduction

The Seals & Identification Laboratory (*SILab*) is part of the Joint Research Centre of the European Commission. *SILab* develops technologies and equipment based on ultrasonic technologies, suitable for sealing and/or identification of nuclear or commercial items. RFID technologies are used to demonstrate the potential of electronic seals and of "smart" containers. As a spin-off of RFID activities, *SILab* is also developing a "SEcure and SAfe MObility NETwork" to improve independence and mobility for visually impaired people.

Regarding seals for nuclear applications, *SILab* has years' experience with ultrasonic seals and equipments for underwater applications, used by both nuclear control agencies (IAEA and DG-TREN/ESO) in Sellafield (UK) and La Hague (F) installations.

CANDU® Reactors are manufactured by AECL (Atomic Energy of Canada Limited) and are continuously loaded (and unloaded) with fresh fuel. The exploitation license at Cernavoda requires that spent fuel bundle stacking frames be sealed. AECL supplied the ARC seal (AECL Random Coil seals) for this purpose.

On IAEA request, a study of the application of *SILab* ultrasonic seals to replace ARC seals for Cernavoda began in 2005. *SILab* ultrasonic seals present, as main advantage, stability against time and radiation. Being purely static pieces of stainless steal they will last the life time of the stacking frames and remain stable as identity. Only the reading equipment need to be maintained.

2 JRC Ultrasonic seal for the underwater CANDU spent fuel storage.

A Candu reactor operates in continuous mode. New fuel is loaded and spent fuel is unloaded every day (up to 16 fuel bundles by day). The spent fuel is then carried to the spent fuel bay where it is placed in stacking frames. Once the frame is full, a frame cover and two seals are applied. The spent fuel will remain some years (typically 7 years) in the spent fuel bay until it can be stored in dry storage. AECL Random Coil seals (ARC seals) are used to seal the stacking frames.

2.1 Specifications

The agency expressed the need to develop a seal that can be used in place of existing ARC seals. The seal must be read with no limitation of time between two readings, giving the same result. It must also use the existing handling tools developed for the ARC seals.

When the frame cover is in place closing a stacking frame, two seals are attached by tightening them on two tie-rods. The tie-rods are fixed at the bottom of the structure of the bay.

The following photo shows the cover of the stack frame with its two seals already attached:

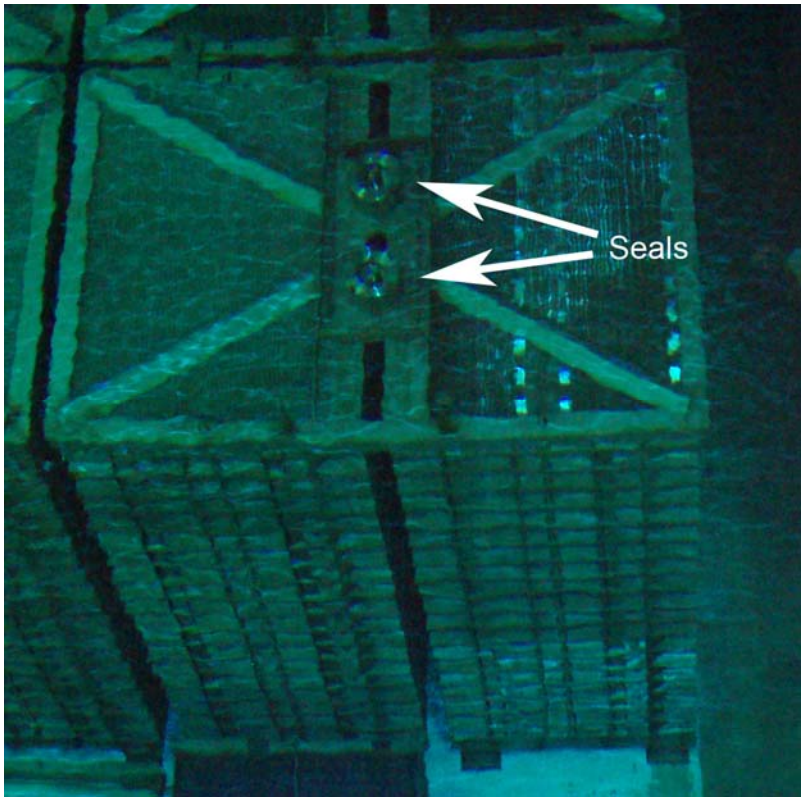

Photo 1: Candu® stack frame.

2.2 The JRC Candu Seal (JCS)

The JCS is based on the *SILab* ultrasonic core, containing both a unique identity (for each seal) and a frangible element (integrity). The identity is a steal piece including a random series of holes and defects inside the metal.

The reading device contains an ultrasonic transducer which generates an ultrasonic signal and senses the reflected signal. The transducer rotates above the seal recording the ultrasonic echoes reflected over a complete revolution. During its revolution it goes also over an integrity link allowing verification of its integrity status.

When the seal is attached, the integrity element must completely engage into the grip of the tie rod as shown in the figure below.

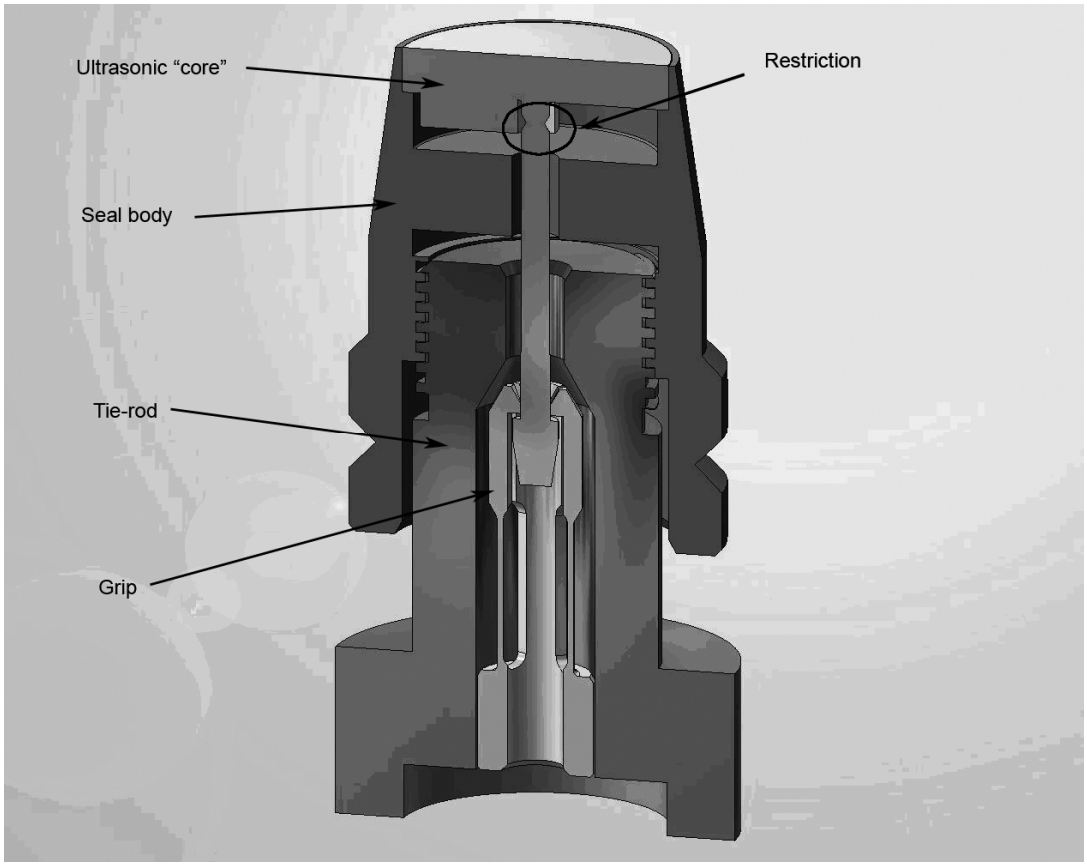

Figure 1: Seal applied in the tie-rod

When the seal is removed, the integrity element is retained by the grip. The integrity element breaks at the level of the restriction. It then falls into the tie rod where it remains. The identity feature remains unchanged.

When a broken seal is read the reading system will detect the absence of the integrity feature but will still be able to check the identity.

2.3 The Reading System

A new Reading Head was specifically designed for the reading of JCS seals. The design provides the ability to check the identity and integrity with a single measurement. The mechanical interface on the top is identical to AECL reading heads to use the same tools for the handling.

Photo 2: JCS reading system

The inspection software was conceived to facilitate "on site" inspections. The first menu level contains only the functions an inspector will have to perform inside the pond. These functions are the reference, attach, verify and detach of a seal. Management functions are accessible from a second level menu. When measuring seals, the relevant parameters are displayed. These parameters are also saved with the measurement and shown when an existing measurement is displayed or for further investigation.

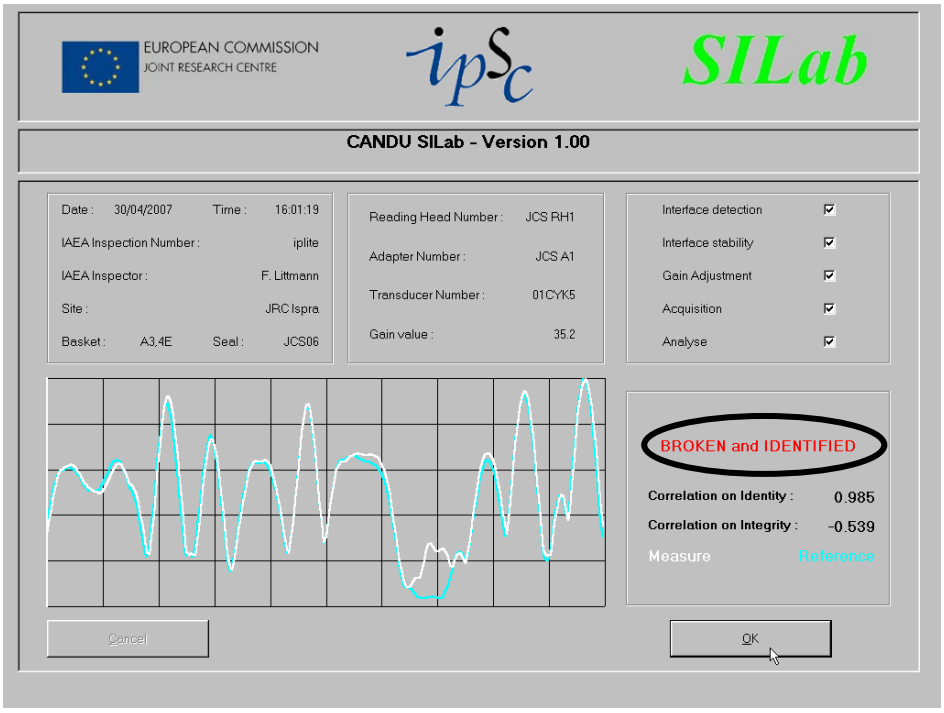

Photo 3: Reading screen of the JCS software

After any acquisition, the new reading is compared to the reference reading and the status of the comparison is indicated together with the correlation coefficient. The status of the seal is indicated in green when the result is "as expected", and in red when not as expected. The example above shows an unexpected broken seal.

3 JCSS Development

The developments of the JCSS system start in mid 2005. The idea was to develop a substitute to ARC sealing systems that can reuse most of the tools used at Cernavoda (Romania), and that does not require regularity in the frequency of inspections.

3.1 Mechanical study of the seal itself

First part of the JCSS study regards the seal itself. The overall dimensions of the seal are two to three time bigger than the previous ultrasonic seals developed at JRC. This has led to different mechanical concepts for the ultrasonic core.

A first attempt was to make ultrasonic identity that covers all the upper part of the seal itself. This creates two main problems. First of all, the disks used for the identity must be flat in order to be brazed. This means to have thick disks, with the consequence that the cavities are bigger and easier to reproduce. The second problem is also a consequence of the overall dimensions of the disks; they required more time to come at good temperature for the brazing process. And some components of the brazing paste will have evaporated before all the identity becomes in temperature. This led to badly brazed identities that can be damaged with shocks.

The picture below shows such identities:

Photo 4: Seal identity using large disks.

A second type of identity was designed that covers only part of the top of the seal, from the center to the side. This permits to have smaller disks and identities. The time require to have all pieces at the brazing temperature is compatible with the volatility of the components of the brazing paste. The photo below shows a first design of the different elements of these identities.

Photo 5: Smaller identity

This configuration still present a problem, adding the integrity link to the upper part of the identity (the so called "delay line") creates a discontinuity in the identity that makes difficult the measure of the integrity. The ultrasonic beam has a diameter on the top surface of the seal that is equal or bigger than the diameter of the integrity link. This results in low signal, and so possible errors in detecting the signal of a broken seal, but also in easy replacement of the integrity link by badly intentioned people (no mechanical continuity required, only a welding point is OK). The solution consists in integrating the integrity link and the delay line in a unique piece by machining a single steel bar. The figure below shows the two configurations. Left is the integrity link that is added to the delay line, right is the unique piece (with already the identity disks brazed).

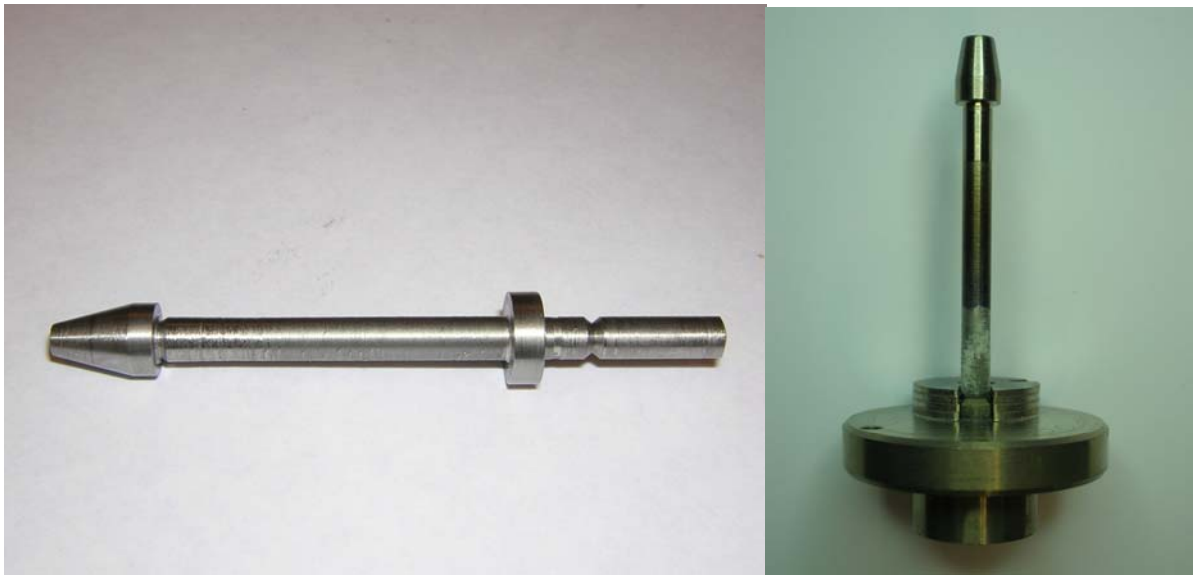

Photo 6: Two solutions for the integrity link

This seal design phase ended toward February 2006 when a first batch of seals was supplied to the IAEA for field tests in Cernavoda (Romania). From this date on, no major modifications were made on the seal design, nor on the ultrasonic core, just a few on the mechanical fitting of the seal onto the tie-rod thread.

3.2 Field test on the first batch of seals, February 2006 - June 2006

The first batch of seals was delivered to the IAEA in February 2006. They were read at IAEA head-quarters in Vienna and then installed at Cernavoda (Romania) in March 2006. The first verification of these seals was accomplished in June 2006.

Special software was developed for these tests. The correlations between Vienna readings and Cernavoda's give very good results (higher than 0.9). The same reading head was used for both readings. Special test software was used for these tests.

A seal was then broken and read again. The figure below shows the response of the test system:

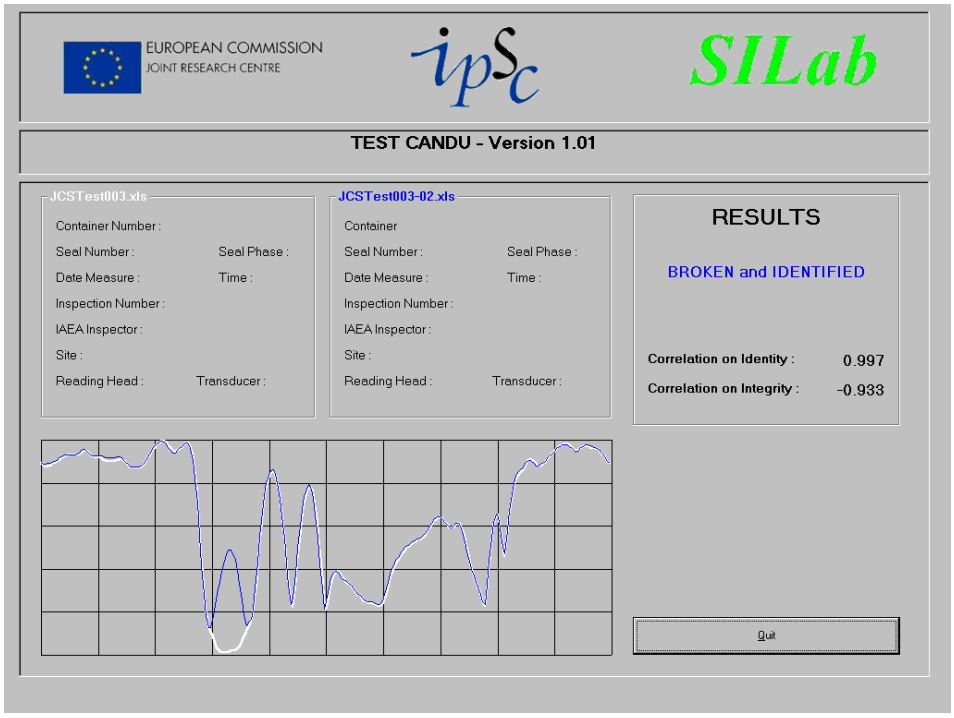

Photo 7: Reading of a broken seal in Cernavoda

3.3 IAEA tests in Cernavoda, June 2006 - September 2007

A second batch of 8 seals was supplied in June 2006. A few minor design modifications were made to improve the fitting onto the tie-rod and facilitate the use of the seals.

The seals were read the first time at Ispra with a new reading-head. They were then read again in Cernavoda with a second reading-head. The correlations between the two readings were all higher than 0.98. This demonstrates that the reading heads are interchange.

The June 2006 tests were done with both teams, SILab's and IAEA's. From that date on and for more than one year, the system was used by IAEA inspectors alone.

During this year of field tests, the inspectors attached the 8 seals and read them several times using the test software.

3.4 IAEA tests in Cernavoda, October 2007 - December 2008

In parallel with the Vulnerability Assessment (VA) done by Sandia National Laboratory (see next chapter), the IAEA decided to substitute half of the ARC seals with JRC seals.

As the stack cover is secured with two seals, and as it is not possible to access the content of the stack without removing the two seals, it remains safe, from a safeguard point of view to substitute one of the two ARC seals with a JCSS one. The continuity of knowledge is assured by the remaining ARC seal on each stack. This gives also the opportunity to have quickly a return of experience from the inspectors on the use of the system and to propose improvements that could be tested by Sandia during the VA.

In September 2007, we presented in Vienna HQ the use of the system to the inspectors that could be involved in Cernavoda operations. This presentation raised several potential improvements that were taken into account in the inspection software (version 1.01) before the ARC seal substitution campaign.

This campaign was done in October 2007 and more than 30 seals were substituted at that time.

Photo 8: Substitution campaign in Cernavoda, October 2007

Following this campaign, IAEA made several inspections at Cernavoda, performing attachments, verifications and detachments of seals. Each new stack was sealed with an ARC seal and a JCSS one. During all these measures, some minor problems appeared due to some weaknesses in the software. These bugs were corrected in the version 1.02 available from January 2009.

3.5 Vulnerability Assessment

A batch of fifty seals was produced to support a Vulnerability Assessment. The IAEA requires a 3rd party Vulnerability Assessment before a new type of seal can be authorized for safeguards use. A new reading head and inspection software were also produced for this assessment.

All seals were read once with our laboratory reading head and then a second time with the reading head produced for the VA. The correlations between these two sets of readings confirmed the results obtained from the second batch of seals. The median of these 50 correlations is 0.983. The measurements were made using the new inspection software (version 1.00 from August 2007).

The 50 seals, the reading system and the inspection software were supplied in August 2007 to the IAEA in Vienna. The kick-off meeting with Sandia National Laboratory was held in Ispra in January 2008. The software was upgraded following inspectors' requests in the meantime.

The results of the VA are not yet public, but in December 2008 the IAEA classified the JCSS sealing system in category A (available for operation).

3.6 Substitution campaign, January 2009

Following the classification of JCSS sealing system in category A, the substitution of the remaining ARC seals by JCSS ones has been done in January 2009. The version 1.02 of the inspection software, correcting the bugs revealed by the inspectors was also installed at that time.

During the substitution campaign, all operations have gone smoothly and the continuity of knowledge in the spent fuel bay of Cernavoda unit 1 is now done by the JCSS system.

4 Other developments of JCSS sealing system

4.1 Cernavoda Unit 2

In 2007, the unit 2 of Cernavoda become operational. The first spent fuel bundle came out in late 2008 and are now stored in their respective spent fuel bay.

Tools, seals and sealing system have been ordered by IAEA to start the sealing operations in 2009. The tools and the seals have already been delivered to the IAEA for Cernavoda unit 2 in January and February 2009 and the starting of operations is expected in the coming months.

Photo 9: Tools and reading system for Cernavoda unit 2

4.2 Kanupp (Pakistan)

Kanupp nuclear plant in Karachi (Pakistan) is also a CANDU reactor. IAEA investigated the sealing of the spent fuel bay. Up to now, there is no lid to close the spent fuel stacks. Together with Pakistani operator, we study a system based on the JCSS sealing system that allow the sealing of these stacks. The figure below shows the last proposed solution:

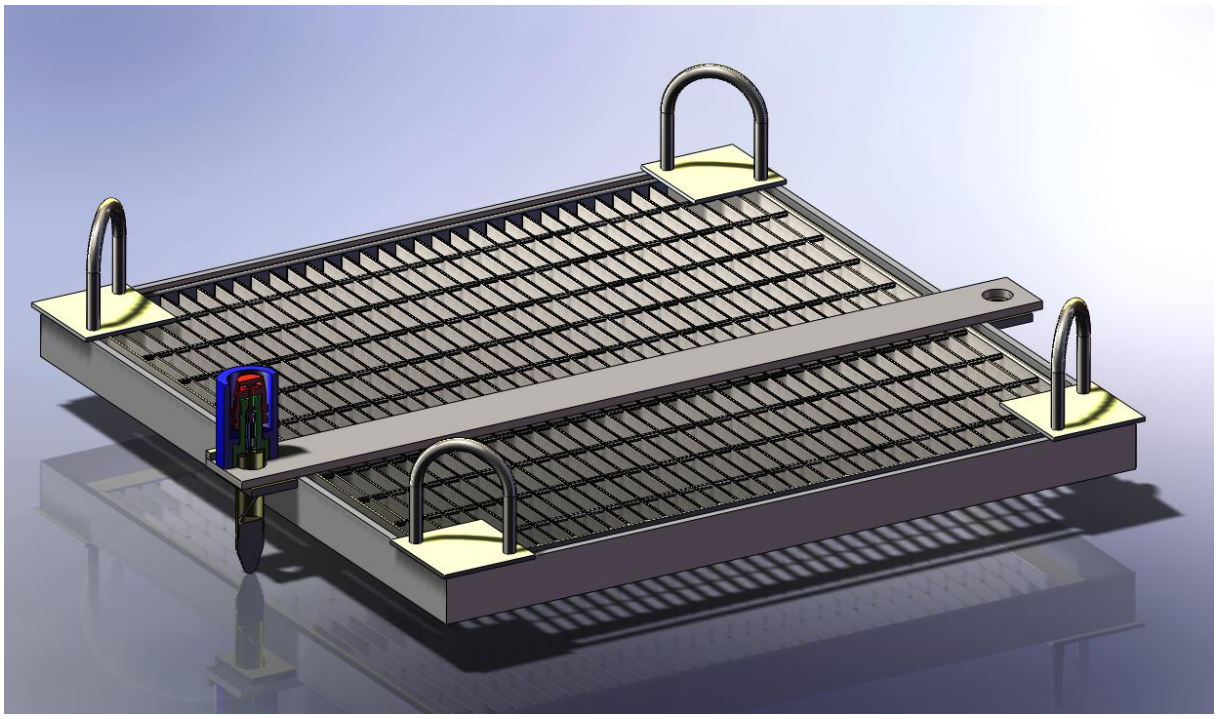

Figure 2: Proposed sealing system for Kanupp spent fuel bay.

The system consists in a grid that closes the top of the stacks. On two opposite sides of the stacks are welded two supports that allow the centring of the grid on top of the stack and on which will be attached the seals. On the figure above, only one seal and its support is shown. Seals and reading equipments will be similar to JCSS ones developed for Cernavoda. The tools will be (slightly) adapted for the Pakistani configuration.

4.3 Constor® container sealing

Together with DG TREN inspectorate, an evolution of JCSS sealing system for the Constor® containers is also under development. Such containers are built in stainless steal and on top of them is placed a concrete lid that act as biological shielding. The request is to seal this concrete lid on top of the container. There is a gap of air between the top of the stainless steel container and the biological protection to allow dissipation of heat. These containers are put in dry storage.

The next figure represents the solution we proposed for this sealing:

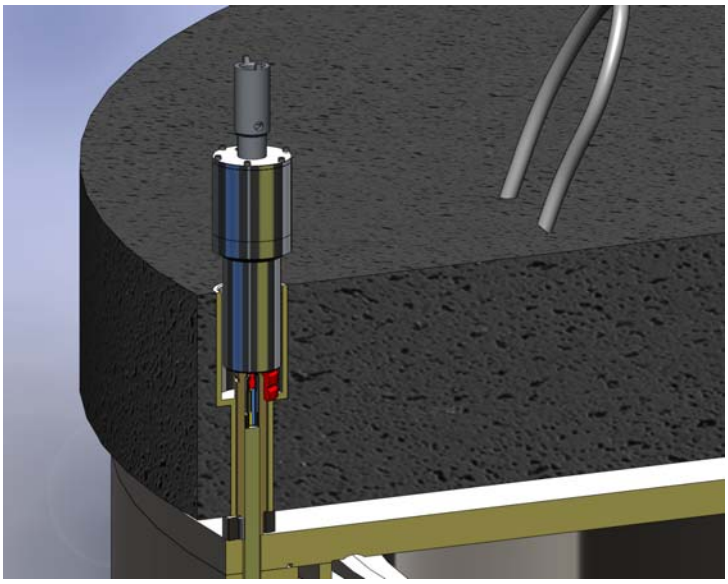

Figure 3: Sealing of Constor® containers (reading head in position upper the seal)

In the steel lid of the container is fixed a small tie-rod that pass thru the separator between the container steel lid and the concrete biological protection. The concrete lids are built with a specific insert in which the tie-rod will enter. Once the lid is in place, a JCSS seal (in red in the above figure) can be applied on the tie-rod making impossible to hold the lid without withdrawing previously the seal. The seal is based on the JCSS approved seal with only minor changes. The reading head will be adapted for the use in dry condition. Water will be necessary for the ultrasonic beam coupling, and so an external tin will bring the water when required.

5 Conclusion

SILab developed in these last years a new seal concept (JCSS), based on ultrasonic detection of defects in stainless steel pieces, for the replacement of the ARC seals in Cernavoda spent fuel bay (Romania). JCSS seals can handle very high levels of radiation, long life time, and in-situ verification.
Several test campaigns were done to test the seal concept and its utilisation together with IAEA inspectors. A third party Vulnerability Assessment is also performed by an external laboratory. As results of all these tests, IAEA decides in December 2008 to classify the JCSS sealing system as category A, available for operation.
The implementation of JCSS to Cernavoda unit 1 has been completed successfully in January 2009. Implementation in Cernavoda unit 2 will start later on in 2009.
New developments of JCSS sealing systems are ongoing, based on the same system (Kanupp spent fuel bay (Pakistan) and Constor® containers).

SESSION 5

SAFEGUARDS CONCEPTS

A NEXT GENERATION SAFEGUARDS INITIATIVE: OUTLINE OF INTERNATIONAL COOPERATION

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Dunbar Lockwood, Kasia R. Mendelsohn, Justin K. Reed,
James K. Sprinkle, and Alexander D. Sunshine

Introduction

The Next Generation Safeguards Initiative (NGSI) was launched in 2008 by the U.S. Department of Energy National Nuclear Security Administration to develop the technologies and expertise necessary to strengthen the international safeguards system as its mission evolves over the next 20-25 years. Though its primary focus is revitalizing the U.S. contribution to international safeguards, we recognize that NGSI cannot succeed as a purely domestic effort. Only by combining U.S. technical and scientific resources with those of international partners will the world be able to keep pace with the emerging safeguards challenges.

In order to increase the benefit of safeguards while reducing their burden, NGSI needs to work with international partners such as the IAEA, Euratom, the United Kingdom, and Japan to improve the quality and cost-effectiveness of international safeguards. The September 2008 International Meeting on Next Generation Safeguards brought together government and technical experts from 11 countries and the IAEA to Washington, D.C. to reach a common understanding of the issues to be tackled:

> The participants addressed IAEA safeguards challenges and opportunities in the coming decades, including growing safeguards responsibilities, expanding interest in nuclear power, high-profile investigations, and limitations on available safeguards technology and expertise. The meeting highlighted the critical importance of promoting international cooperation to anticipate challenges and revitalize national capabilities to support the IAEA in its mission to verify peaceful uses of nuclear energy on a continuing and reliable basis.[1]

Following that consensus, NGSI is moving forward to engage with partners across its program elements. This paper presents an overview of those elements, a review of NGSI programmatic implementation and international cooperation, illustrative examples, and a preview of possible next-steps for NGSI international cooperation.

Overview of NGSI Program Elements:

The NGSI program is organized into five program elements:

[1] "Remarks by Conference Chairman Adam Scheinman Next Generation Safeguards Initiative Inaugural Conference, September 11-12, 2008," U.S. Department of Energy National Nuclear Security Administration. Available at http://nnsa.energy.gov/nuclear_nonproliferation/2147.htm

LA-UR-09-03014

(1) Policy Development and Outreach; the goal of this element is to support U.S. safeguards policy development and work bilaterally and multilaterally with other countries to strengthen the international safeguards system;
(2) Advanced Concepts and Approaches; this element will develop advanced safeguards concepts and approaches as well as assessment methodologies that enhance the effectiveness, efficiency, and credibility of international safeguards;
(3) Technology and Analytical Methodology Development; will renew and strengthen the international safeguards technology base, through the application and development of instrumentation, as well as analysis and system tools, that optimize the implementation of safeguards at both the facility and the state levels;
(4) Human Capital Development; the goal of this element is to expand the international safeguards human resource base by attracting and training the next generation of international safeguards experts; and,
(5) Nuclear Safeguards Infrastructure Development; will assist in the development of national infrastructures in all countries that have nuclear power or plans for nuclear power.

NGSI aims to realize its goals by strengthening DOE's primary technical asset, the U.S. National Laboratories, and by engaging industry and academia. Also, NGSI is intended to re-energize the U.S. capability to be a world leader in the international safeguards system. Finally, NGSI is intended to engage and re-energize the global nuclear safeguards community.

While NGSI has a U.S. domestic focus, its underlying purpose is international; we recognize that this initiative cannot succeed as a purely domestic effort. Rather, our effort is intended to serve as a catalyst for a much broader commitment to international safeguards in partnership with the IAEA and other countries. Only by combining U.S. technical and scientific assets with the assets of international partners will our world be able to keep pace with the emerging safeguards challenges.

NGSI Programmatic Implementation and International Cooperation

DOE/NNSA is actively working on projects in each of the five program elements. While some elements are focused primarily on developing U.S. domestic policies and capabilities, all of the areas hold opportunities for international cooperation and engagement.

1) Policy Development and Outreach.

The goal of the Policy Development and Outreach element of the NGSI Program Plan is to "support U.S. safeguards policy development and work bilaterally and multilaterally to strengthen the international safeguards system as an essential element of the global nuclear nonproliferation regime."

In October 2007, NNSA's Office of Nonproliferation and International Security completed a wide-ranging study on the current and emerging challenges facing the

international safeguards system.[2] This report recommended that the United States "work to put into effect the policies and authorities necessary for the IAEA to accomplish its evolving mission, in particular by considering more frequent use of special inspections" and providing assistance to promote the universal adoption of the Additional Protocol (AP).[3]

As part of NGSI's implementation, NNSA has established a Policy Working Group, which has begun new studies to evaluate the IAEA's budget with the understanding that the Agency's responsibilities have been expanding faster than its resources. This "requirements-based" study will consider infrastructure and operating costs and look for opportunities for cost savings that do not diminish safeguards effectiveness, while remaining mindful of the IAEA's request for substantial budget increases. In addition, the Working Group has initiated a study on the IAEA's legal authorities that will examine, inter alia, lessons learned from past non-compliance cases, and what can be done after a state withdraws from the Non-Proliferation Treaty (NPT) after violating its obligations.

The Policy Working Group will also assess safeguards enhancements, including opportunities for greater information sharing between Member States and the IAEA, investigation of weaponization and procurement activities, and ways to strengthen the state-level approach to safeguards. Improved coordination of IAEA and Nuclear Supplier Group (NSG) activities in monitoring global nuclear commerce will also be examined.

2) Advanced Concepts and Approaches.

The goal of the NGSI Progam Plan's Concepts and Approaches element is to "develop advanced safeguards concepts and approaches and assessment methodologies to enhance the effectiveness, efficiency, and credibility of international safeguards."

NGSI anticipates the safeguards challenge presented by deployment of new types of reactors and fuel cycle facilities. Also, the limited international safeguards resources that are available will need to be used more effectively and more efficiently, especially in plants that pose the largest burden, namely complex, large, bulk-handling facilities. These challenges will require advanced safeguards concepts and approaches to contain costs while maintaining effective safeguards.

As an early step, NGSI will seek to institutionalize "Safeguards by Design" as a new approach for safeguards. Safeguards by Design is an innovative approach that has potential to advance the safeguards "state of the art" in new facilities by incorporating early modeling and analysis of facility process flows, and integration of advanced measurement instrumentation and monitoring systems into facility design and construction. This approach is intended to ensure that international safeguards

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[2] U.S. Department of Energy National Nuclear Security Administration, "International Safeguards: Challenges and Opportunities for the 21st Century." Available at http://nnsa.energy.gov/nuclear_nonproliferation/documents/NGSI_Report.pdf

[3] The U.S. Additional Protocol (AP) entered into force on January 6, 2009. The United States will work with the IAEA and others to promote universal adoption of the AP.

requirements are fully integrated with safety and operational considerations from the outset of the design process of a new nuclear facility. Safeguards by Design, which is also intended to identify facility design features that would facilitate safeguards implementation, could lead to efficiencies and could help avoid costly and time-consuming retrofits. Preliminary studies are underway, and we hope to expand Safeguards by Design into a formal, multi-year project that would eventually become a universally applied standard for new nuclear facilities. We propose to broaden the meaning given to Safeguards by Design. There is an important difference between designing in safeguards from the start and designing the facility – its material flows and containment features – to make it easier and less costly to safeguard. In order to advance this process, we plan to work with the IAEA and others to convene an international working group to establish criteria, best practices, and design guidelines. Then we must work globally with states and vendors to put these design practices into effect.

The NGSI program has sponsored presentations on Safeguards by Design by national laboratory experts at the IAEA's "Workshop on Facility Design and Plant Operation Features that Facilitate the Implementation of IAEA Safeguards" in October 2008, and at the "International Symposium on Nuclear Security" in March/April 2009. NNSA has also been engaging regularly with private industry in the United States on Safeguards by Design issues.

Updating the safeguards approach for gas centrifuge enrichment plants; new large-scale commercial reprocessing plants; and CANDU and Pebble Bed reactors are other priorities of this effort. We will engage the IAEA and other states on new safeguards approaches, and help develop guidance documents for strengthening safeguards activities such as DIV and process monitoring.

Advanced process monitoring approaches are also being explored under NGSI. Consistent with the IAEA's practice of utilizing all available information in drawing safeguards conclusions, we are examining how a wide range of operations data might be used to better safeguard nuclear facilities. This approach goes beyond acquiring data necessary to support accurate material accounting and seeks to verify the operational history of the facility. Much work is needed for example to develop data requirements, and address authentication and proprietary concerns, but such techniques could increase both the effectiveness and efficiency of safeguards at complex facilities.

3) Technical and Analytical Methodology Development.

In its first year, NGSI has begun to investigate a number of new non-destructive assay technologies, and in particular has initiated a major project to conduct an integrated assessment of technologies with the potential to directly quantify plutonium content in spent fuel. Early efforts have focused on modeling these technologies against a library of virtual spent fuel assemblies. The most promising technologies will be selected for further development, and we hope to work with our international partners to test some of these technologies on actual spent fuel assemblies in the future.

A workshop on new types of detector materials will be held at LLNL, starting in late June. The results will be published. Additional projects have been initiated in data integration, containment and surveillance, data authentication, and environmental analysis. Many of these projects will benefit from international cooperation. A survey of U.S. reference materials and of U.S. laboratory and infrastructure needs has been initiated in view of the downsizing and reconfiguring of the U.S. defense programs and the corresponding facilities. This survey may highlight the benefits of or need for international collaborations to gain access to operating facilities for development of instruments and safeguards approaches. The survey should be completed in 2009 and as a consequence, cooperation could begin to grow as soon as 2010.

NGSI will also expand programs by the Office of Nonproliferation and International Security to conduct joint safeguards technology development focused on resolving facility-specific safeguards challenges through bilateral safeguards cooperation agreements with our international partners. The U.S. Department of Energy and the European Community through Euratom have a long history of such cooperation. Since signing the newest Nuclear Material Safeguards Research and Development Agreement in 1995, the two parties have entered into 35 cooperative activities. DOE-Euratom collaborations have covered a wide range of safeguards topics including detector characterization, radiation signature simulation, data authentication and security, and plutonium measurements. DOE-Euratom activities tend to focus on measurement data analysis, data fidelity, and system characterization. These areas of emphasis are unique to this agreement and generally serve broad international safeguards issues in addition to issues particular to European safeguards inspection efforts. NGSI is expected to increase and encourage these collaborations.

There are currently nine cooperative safeguards projects underway with Euratom and three more soon to begin. Ongoing projects include the development and delivery of a simulator for training inspectors in interpreting plutonium gamma-ray spectroscopic data, improvements to the International Neutron Coincidence Counter software (INCC) for neutron measurement analysis, integration of the Origen-Arp code into the Euratom RADAR/CRISP codes, investigations of new safeguards data authentication techniques, and a tool that combines a 3D laser scanner with gamma imaging to improve nuclear material holdup measurements.

DOE's Office of Nonproliferation and International Security has a number of similar relationships with other international partners. We have had successful collaborations with Japan on plutonium holdup measurements and reactor gate monitors, with ABACC on gamma spectroscopy analysis and 3D-DIV, and with the Republic of Korea on Laser Induced Breakdown Spectroscopy and Cm/Pu ratio monitoring to mention just a few examples. We hope to expand on these collaborations with new projects in areas of mutual interest including process monitoring, data integration, and equipment testing and demonstration. In addition, we hope to conclude agreements with new countries interested in broadening their familiarity with and support for international safeguards.

4) Human Capital Development.

The goal of the NGSI Program Plan's Human Capital Development element is to "revitalize and expand the international human capital base" by attracting, recruiting, and training a new generation of international safeguards experts.

To address the looming human capital crisis, NGSI is taking steps to revitalize and expand the human capital base, with programs to cover the full spectrum of current and emerging safeguard-relevant disciplines. We have taken a number of initial steps to implement our Action Plan to develop and educate the next generation of U.S. international safeguards specialists. We initiated two new summer courses on international safeguards issues and nonproliferation in 2008 through national lab-university partnerships, and are adding a third course, on safeguards policy, in the summer of 2009.[4] We increased funding for summer student interns in international safeguards at our national laboratories in 2008, and hope to expand this program by 50 percent to 75 interns in 2009. In addition, the NGSI program will sponsor postdoctoral fellows in international safeguards R&D at four (or more) national laboratories this year. We are also initiating a number of new lab-university collaborations through which national lab-based international safeguards experts will work with university faculty in developing graduate level courses on international safeguards and nonproliferation at nine U.S. universities,[5] and plan to expand this, funding permitting, in 2010. As part of this collaboration with universities, we are supporting a workshop in August 2009 for university faculty on safeguards and nonproliferation educational approaches and course design, and hope to provide budgetary support for a number of university faculty-national laboratory joint appointments.

As important as university engagement is to our program, we are also mindful that many safeguards professionals entered the field after they started their careers. Accordingly, we are supporting professional development programs at several U.S. national labs to help attract and introduce early and mid-career professionals into the safeguards field. Lastly, we are following up to a conference in October 2008 on issues we face in recruiting strong U.S. candidates for safeguards positions at the IAEA, and then facilitating their reentry into the U.S. workforce when they return, with an enhanced program to recruit and prepare U.S. candidates for safeguards employment at the IAEA.

Complementing these U.S.-focused efforts are a number of activities for engaging with our international partners in addressing what are common challenges in developing the next generation of international safeguards experts. Training and education, and in particular regional programs and exchanges of experts, practitioners, and students will play an important role in building safeguards expertise. We are considering ways to

[4] In a manifestation of strong student interest in the international safeguards field almost 100 students applied for approximately 60 total spots in the two summer safeguards policy courses.

[5] The nine universities include the University of Florida, University of Michigan, University of Tennessee, University of New Mexico, North Carolina State, Georgia Tech, Washington State, Oregon State, and Washington University.

encourage regional groupings of countries to serve as clearinghouses for information, training materials and cooperation.

At the September 2008 International Meeting on Next Generation Safeguards in Washington, participants agreed that varying levels of technical sophistication across states poses a challenge in developing a common human capital development approach. Therefore, regional programs and personnel exchanges focused on training and education will play an important role in building safeguards expertise. We are considering ways to encourage regional groupings of countries to serve as clearinghouses for information, training materials and cooperation.

The September 2008 meeting recommended a follow-on NGSI Human Capital Development Workshop. This workshop, which will be co-hosted by Euratom is still under development. It will likely take place in Ispra, Italy in September 2009. Tentatively, the workshop will focus on three areas: (1) developing common curricula and training materials for safeguards; (2) discussion of possible joint training activities and facilities; and (3) discussion of potential exchange programs.

5) Nuclear Safeguards Infrastructure Development.

NGSI will work with the IAEA and international partners to develop strong nuclear safeguards infrastructure, especially among states with limited nuclear power programs or those expressing interest in such programs. This element of NGSI will dovetail with the established IAEA Milestones process and the linking of safeguards with safety and security, as set forth in the 3S's concept introduced by Japan and endorsed by the G-8.

Just as the nuclear industry has developed a "safety culture," we will seek to develop a "safeguards culture." Through NGSI, we will work with our international partners and industry to demonstrate that nuclear safeguards are not a burden to endure, but rather a means to ensure the reliability, safety, and security of nuclear energy, and ultimately good safeguards is in the best interest of both states and industry.

Last September, the NGSI International Meeting participants emphasized the need for coordination among states that provide assistance, as well as those that receive it, to ensure consistency of message and goals and to avoid duplication of effort. Suggested actions included a resource survey in coordination with the IAEA to determine needs, development of standardized guidance for national legislation and training materials, and organizing assistance on a regional basis.

In pursuit of that goal, we will host an NGSI Infrastructure Outreach Harmonization Workshop this June in Vienna. The workshop will be an opportunity for organizations like Euratom to exchange information on various assistance, training, and outreach efforts related to international safeguards. We hope this workshop will improve international coordination of assistance, and serve as a way to exchange views on the role of safeguards in the development of infrastructure for nuclear power. Areas in which we particularly hope to coordinate our efforts include State System of Accounting and

Control (SSAC) training, Additional Protocol implementation assistance, regulatory development assistance, and human resource development.

During this past year, NGSI has conducted bi-lateral training addressing enrichment plant safeguards (Argentina), fuel fabrication safeguards (Indonesia) and fundamentals of nondestructive (South Africa) and destructive assay (ABACC) techniques. A core part of the cooperative program works with other countries to solve very specific safeguards technology needs, such as material measurements at Chernobyl nuclear power plant (Ukraine).

Next Steps In NGSI International Cooperation

Planning for a late September 2009 follow-on meeting to the 2008 NGSI International Meeting is in the early stages. This meeting will call together safeguards experts from around the world to exchange information about safeguards related activities to enhance cooperation and reduce any chances of duplication. Participants will be encouraged identify areas of possible cooperation to strengthen the international safeguards system.

NGSI's success will be determined by our ability to attract partners and promote collaboration. Many U.S. facilities were built for nuclear weapons programs, and are now shared by multiple users. As the weapons programs downsize and the complex is reconfigured, U.S. safeguards programs are facing the loss of experimental facilities and nuclear materials. We are considering ways to further share facilities in other countries to test safeguards technologies and techniques, to share research and field trials, to engage industries and technical communities, to promote information exchanges and best practices, and to work together to ensure safeguards authorities are used to their fullest.

International safeguards challenges will only increase if they are left unaddressed. We recognize that U.S. participation is critical in bringing together international partners in a comprehensive and concerted effort to revitalize international safeguards. However, we acknowledge that the United States can not accomplish its goals without full and open partnership with other states. International safeguards as well as regional safeguards must have the best staff, technology, operations and methodologies available, as well as adequate budgetary support. International efforts should be clearly focused on this goal of providing safeguards with the tools and resources it needs to accomplish its mission.

We believe that, if we pool our resources and work together, we can improve the effectiveness and the efficiency of international safeguards -- a goal that is more important than ever if the international community is going to enter the anticipated nuclear renaissance without increased risks of proliferation. Only by combining U.S. technical and scientific resources with the resources of international partners will our world be able to keep pace with the emerging safeguards challenges.

The Fissile Material Cut-off Treaty: An Historical & Problematic approach

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***Abstract**: The idea of a treaty or a Convention banning the production of fissile material for the nuclear weapons or other nuclear explosives devices (also known as Cut-off treaty) to cap the development of nuclear weapon arsenal is not a recent one. This idea was floated around early after the end of the Second World War when the government of the United States emerged the needs of such a ban. Since that time, the idea of a Cut-off treaty has experienced various fortunes, sometimes at the top of disarmament agenda, sometimes consigned behind the scene. Since a couple of years, a Cut-off treaty seems to be more within the reach of the international community and several states have called again for the commencement of negotiation and the trend of the work at the Disarmament Conference (CD) in Geneva has revived some hope that this negotiation may eventually start. The need for a didactic article relating the complex history of the discussions for Cut-off treaty and describing the issues associated which such a treaty has been raised by the members of the Verification Technologies and Methodologies Working Group (VTM) of the ESARDA during their meetings. This lecture and the associated article to be published in the ESARDA Bulletin attempts to fulfil this need in a neutral and impartial but as comprehensive and understandable as possible way. It first relate the history of a treaty on a ban of the production of fissile material for nuclear weapons and other nuclear explosive devices and its links with other non proliferation and disarmament instruments,. It gives the declared position of major actors on the treaty from the origin to the current status. Then the article address the elements of the treaty, its possible objectives and scope, the definitions in particular the definitions of fissile material for nuclear weapon and their production, the issue of activities non prohibited under the treaty, the problematic of a verification system and the issue of compliance. It should be noted that the views expressed in the lecture and the associated article are those of the author and do not necessarily reflect the views of the CEA nor the French Authorities.*

Keywords: FMCT; Cutoff treaty; fissile materials; nuclear weapons; verification

The full article will be published in the ESARDA Bulletin.

Safeguards Implementation at Small Enterprises: Radiation Safety Inspection and Accounting of Industrial Radiography Containers

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Abstract

The accounting for industrial radiography containers made of depleted uranium poses special challenges to the safeguards regime.

Depleted uranium is counted as nuclear material and is as such under safeguards, however, according to the comprehensive safeguards agreement it can be exempted from safeguards under certain conditions. The implementation of the Additional Protocol opens for Agency inspections even of the exempted material. The usefulness of the exemption of material that is readily transportable and subject to possible inspections anyway is discussed.

The question of whether the locations of this material should be declared as sites or not is treated. The rather detailed requirements to a site declaration and the usefulness of these in this context constitute the background for this treatment. Connected to this is the structure of MBAs and KMPs, the Agency accountancy system does not allow more than 26 KMPs under one MBA due to restrictions on the number of characters in the KMP numbering. A pragmatic solution to this dilemma is presented.

The volatility of the inventory of shielding containers due to frequent changes in company structure is also discussed as a challenge to the system both from the point of view of site declaration and exemption.

The combination of safeguards physical inventory takings and ordinary radiation safety inspections of the radiography company as a means to ease the work load both for the authorities and for the companies is also elucidated and experiences described.

The paper concludes with an evaluation of the risk of proliferation of depleted uranium in shielding containers and the need to put effort in the accountancy of such containers.

Keywords
Depleted uranium, shielding containers, Additional Protocol, accountancy, proliferation risk.

1. Introduction

In the comprehensive safeguards agreements between a State Party and The International Atomic Energy Agency (INFCIRC/153, 177, 193 and others) [1], depleted uranium is counted as nuclear material and is as such under safeguards. Pursuant to article 37, however, it can be exempted from safeguards under certain conditions. If it is to be exported, or the accumulated amount increase above the specified levels, safeguards has to be reapplied. Under such circumstances, the State Party has to keep track of the exempted material to be able to decide if or when safeguards is to be reapplied.

The implementation of the Additional Protocol [2] to the Safeguards Agreement article 2 a. vii) requires the State Party to declare the exempted material, and article 4 a. i) of the AP opens for Agency inspections of the exempted material.

2. Accountancy of DU in industrial radiography containers

The implementation of the Additional Protocol puts further obligations on the State Party to keep track of the depleted uranium. This involves the accountancy and control with small enterprises all around the country, an issue which has been raised at ESARDA meetings before [3,4].

The Norwegian State System for Accountancy Control (SSAC) has therefore questioned the usefulness of the exemption from safeguards of this material when the SSAC has to maintain a comprehensive oversight of the material. It will not make any difference in effort for the SSAC to exempt it. The material and its location will be subject to possible inspections either under the Safeguards Agreement if not exempted under article 37, or as a complementary access under the AP if exempted under article 37.

The Norwegian SSAC considers that the best solution is not to exempt it. This option will keep the inventory continuously visible in the accounts of the Agency. This enhances the transparency of the system even though the Agency would prefer an exemption.

3. Declaration as site or location outside facility

Should the locations of this material be declared as sites or not? There are rather detailed requirements to a site declaration and the usefulness of these in declaring an industrial radiography company is questionable.

We have chosen not to declare them as sites. The containers are often moving around from one workplace to another with a central storage area where rarely half of the inventory is present during working hours. For companies working for the oil industry, the industrial radiography sources may be in use on platforms offshore for weeks. Some of the companies are even one-person enterprises having one industrial radiography container stored in the basement of their home.

We do not consider any of these places as places where proliferation of nuclear material for weapon purposes is going to take place. We also consider the requirements for a site declaration as too detailed for such locations, as this will require maps and drawings of the places and buildings involved. This information is often private or proprietary for economical reasons and it also tends to change frequently. The workload will be out of proportion to the security gain.

4. Organisation of a MBA for small users

The organisation of the MBA to which the small users belong has been a puzzle. We have explored the possibility of giving each of the industrial radiography containers their own KMP under the MBA NO-D for small users in Norway.

Along the way we discovered that the Agency accountancy system does not allow more than 26 KMPs under one MBA due to restrictions on the number of characters in the KMP numbering(!). Hence, this solution turned out not to be feasible.

The pragmatic solution to this dilemma was to include all the companies as a list of addresses under one KMP. That gives a detailed list of locations that can be tracked by a GPS-system or other mapping services available on the Internet. It should also ease the burden of keeping track of the containers when they are transferred between the companies for hire, lending etc. as they are all in one KMP. The permanent transfers, however, are registered in our state accountancy system.

5. Inspections

The Norwegian Radiation Protection Authority has both the function as a regulatory body for utilisation of ionising radiation and the function as SSAC. This creates the possibility of combination of

safeguards physical inventory takings and ordinary radiation safety inspections of the radiography company.

The safety with work operations, manuals, approval certificates, management systems etc. is inspected. By adding the control of type, serial number and weight of depleted uranium as a point in the radiation safety inspection check list both purposes are reached with minimal effort.

This combination has also lead to an increase in inspection activity as more personnel are available to conduct inspections. The combined inspection activity means better radiation safety and better accountancy for nuclear material, a win-win situation for the authorities.

6. Proliferation issues

The last question we have asked ourselves is why we are doing all this accountancy for depleted uranium as shielding containers for radioactive sources [5]. We all know that depleted uranium is in use for a long range of different purposes, both civilian and military. Some of these purposes, such as the use as counterweights pose possibilities for proliferation. Depleted uranium is also found on several battlefields around the world.

The way from depleted uranium to a nuclear weapon is very long, and history has shown that neither the route through enrichment nor the route through breeding of plutonium have lead to a nuclear device based on depleted uranium. Natural uranium from mines has so far been the source material in such attempts.

The puzzling question will always be: Will this partly control of depleted uranium make the world safer, or would intensified control over other uranium products be a better way? And as a consequence the control of depleted uranium as shielding containers and other commercial items could be eased.

7. Conclusions

The combination of radiation safety inspections and safeguards inspections has a potential for enhanced safety and security with minimal effort. However, the need for neatly accountancy with depleted uranium needs to be revisited.

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Proliferation risks of highly enriched uranium (HEU) used for medical isotope production

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Abstract:

To assess the proliferation threat from medical isotope production, the amount of required HEU has been estimated by analysing the world wide consumption of the most common medical isotope, Tc-99m, and calculating the related irradiation procedure. The results of this method are compared with other estimations and the influence of different parameters is analysed, showing that the HEU use for Tc-99m production most likely amounts to about 15 kg per year. The lower estimate is 10 kg/a, the upper bound is 100 kg/a. Most of the excess uranium undergoes liquid storage awaiting its disposal as waste instead of recycling, posing a proliferation threat.

Keywords: Molybdenum, technetium-99m, isotope production, HEU, proliferation

1. Introduction

The use of radioisotopes for medical procedures has significantly increased for the last two decades as therapeutic and diagnostic procedures are being refined and more and more people have access to them. Most medical isotopes are produced in reactors by neutron irradiation of highly enriched uranium (HEU). To assess the resulting proliferation threat, one needs information on the amount of HEU that is used, however this information is not disclosed by the isotope producers.

This paper will try to calculate the HEU requirement based on the actual consumption of medical isotopes. To this end, it will concentrate on Tc-99m, an isotope which is commonly used for examinations and accounts for about 80% of all radioisotope applications in medicine. It is produced and delivered in form of its predecessor Mo-99.

	MDS Nordion	IRE Fleurus	Mallinckrodt Medical	Necsa / NTP
Country	Canada	Belgium	Netherlands	South Africa
Reactors for target irradiation	NRU (CDN); Maple I & II (CDN)	BR-2 (NL); Osiris (F); HFR (NL, F)	HFR (NL)	Safari I (ZA)
Share of world demand[1]	40%	20–30%	25 %	10–15%
Production capacity[2]	5000 – 6000 Ci/batch (several batches per week)	10000 Ci/week	10000 Ci/week	8000 Ci/week

Table 1: Major suppliers of Mo-99

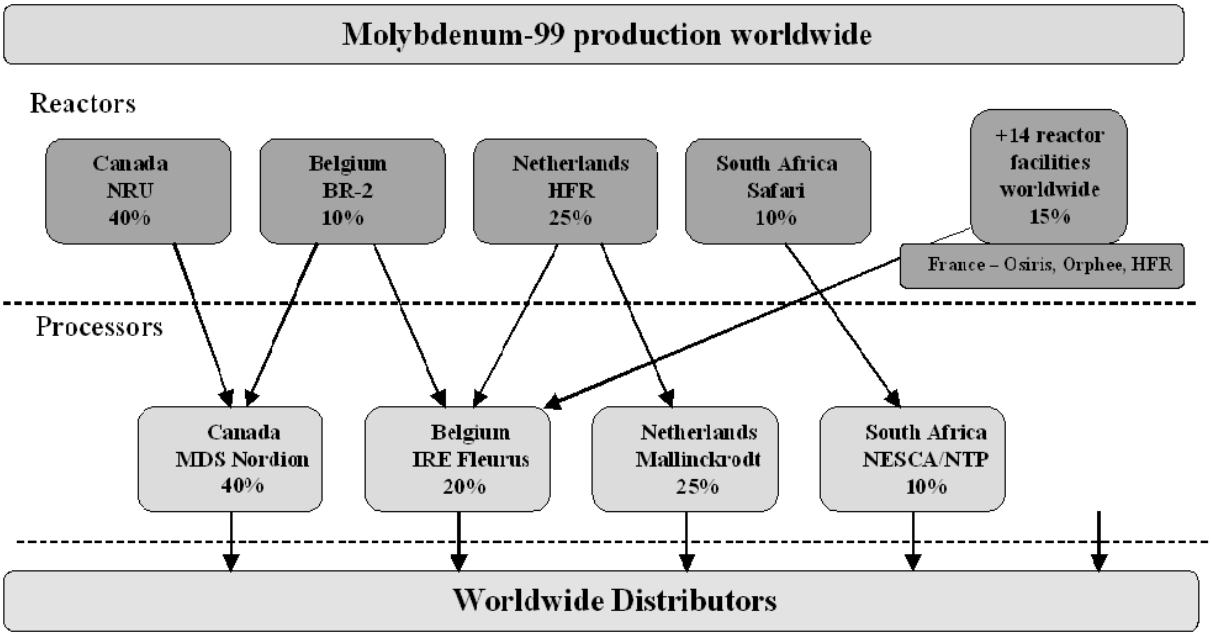

Figure 1: Relation of irradiation reactors and the major processing facilities for global molybdenum-99 production.

The worldwide demand for Mo-99 is almost completely covered by the four largest producers: MDS Nordion, IRE Fleurus, Mallinckrodt Medical and Necsa/NTP [1]. Their production capacities and supplying research reactors are listed in Table 1. The relation between the major isotope extraction facilities and various production reactors is depicted in Figure 1. Currently 95–99% of Mo-99 is produced by irradiation of HEU targets, although the Reduced Enrichment for Research and Test Reactors (RERTR) programme promotes the conversion to LEU since 1978. As a result of these efforts to eliminate the civilian use of HEU, the global HEU demand for research reactors has been declining from 1,400 kg/a in 1978 to 800 kg/a in 2008, and is projected by Reistad & Hustveit [4] to further decline to 500 kg/a in a few years and to around 100 kg/a by 2020. In contrast, the use of HEU for medical isotope production is increasing and may amount to an annual consumption of 100 kg soon.

2. World wide technetium consumption

The worldwide Tc-99m consumption has been estimated by Grosch [5] based on global health care

1 Bonet, David & Ponsard [1] and Ferguson, Kazi & Perera [2]
2 IAEA [3]

data. This method is based on a report by UNSCEAR [6] which had estimated the global annual usage of medical isotopes. To that end, countries were divided into different health care levels defined by the population per physician as seen in Table 2, and the per capita rate of radiopharmaceutical procedures in each health care level. It is assumed by Grosch [5] that, while the absolute amount of procedures has increased since the 1990s, the relative frequency remains unchanged between the different health care levels.

Health care level	Population per physician	Fraction of world population	Usage of Tc-99m	People examined	Total activity administered [TBq/a]
I	< 1000	0.26	0.0256	39,900,000	24,000
II	1000 – 3000	0.53	0.0031	7,850,000	4,700
III	3000 – 10000	0.11	0.000180	118,000	71
IV	> 10000	0.10	0.000009	5,400	3
Total				47,900,000	**28,800**

Table 2: The global usage of Tc-99m by countries of different health care levels.

To obtain an estimation of the current use of radiopharmaceuticals in countries of health care level I, Grosch [5] has analysed the radioisotope use in Germany, Sweden and the United States, the latter of which is often claimed to account for one half of the global Mo-99 demand. Based on these data, one can assume that in health care level I countries, 3.2% of the population receive an examination or treatment using radionuclides in one year. Based on this assumption, the number of radiopharmaceutical procedures in the countries with the highest health care level can be estimated as demonstrated in Table 2. If approximately 80% of the procedures have involved Tc-99m with an average administration of 0.6 GBq Tc-99m per treatment, the total annual activity used in all Level I countries is assessed to be 24 PBq. According to fixed relationships between the four health care levels, the total amount administered worldwide in the last years would be 28.8 PBq of Tc-99m per year. According to UNSCEAR [6] the Tc-99m consumption was about 16 PBq/a in the years 1991–1996, suggesting an increase of 4.5% per year since then.

3. HEU consumption for medical isotope production

Certain generalized assumptions are made to calculate the global HEU consumption needed to deliver sufficient Mo-99 to serve the global Tc-99m demand. These parameters were selected under the assumption that the procedures are optimized and an efficient but realistic balance of HEU consumption per Mo-99 delivery is achieved.

In the reactor, HEU targets of about 4-20 g each are irradiated using thermal neutrons for several days. After irradiation, the targets are dissolved in nitric acid and the Mo-99 extracted. The remaining waste is prepared for final disposal or, in some facilities, for later retrieval of the remaining HEU.

To meet the global Tc-99m demand of 28.8 TBq, a certain amount of Mo-99 has to be produced. As these isotopes continuously decay during storage, processing and transport, the production to meet the global demand depends on the duration of the processing and delivery of molybdenum to the hospital as well as the efficiency of use. At the medical site, Tc-99m is withdrawn from the dispenser on a daily basis, as seen in Figure 2. This withdrawal procedure can be very efficient, extracting 99 % of available Tc-99m.

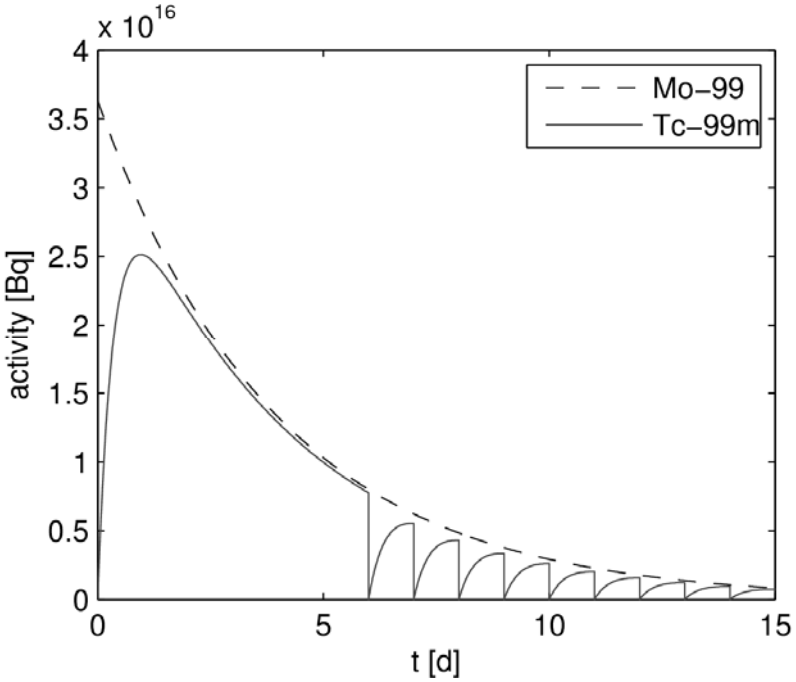

Figure 2: Development of the Mo-99 and Tc-99m activities during a 6 day transport phase and the consequent daily Tc withdrawals at the hospital.

Based on an estimation of the transport time (our assumption: 6 days) and the number of withdrawals (our assumption: six times) one can put the Tc-99m consumption in relation to the amount of Mo-99 produced. The result for this scheme is that about 0.7 TBq Tc-99m are gained for each TBq of Mo-99 produced. According to Table 1, the annual production capacity of the four largest producers is roughly 50,000 6-day-Ci per week, which translates to about 380 PBq/a of Mo-99. Using the factor 0.7 as explained above, one can derive that the main producers have the capacity to provide enough material for a Tc-99m consumption of 270 PBq/a. A comparison to our estimate of 28.8 PBq/a suggests that 10 % of the overall capacity is used, which is plausible.

The required HEU mass can be derived by simulating the irradiation procedure. In the model, targets with 15 g HEU of 93 % enrichment were assumed, exposed to a thermal neutron flux of 10^{14} $s^{-1}cm^{-2}$ for 5 days. Fission yield data from ENDF VI were utilised. The cooling and processing of the irradiated targets is assumed to take about 2 days.

As a result of the calculation using these parameters, approximately 1000 targets of 15 g are needed to meet the global Tc-99m demand. This material contains nearly 14 kg U-235, i.e. a little more than half a significant quantity that the IAEA defines to be 25 kg.

4. Results

The calculations show a global HEU usage of about 15 kg per year for medical isotope production. The International Panel on Fissile Materials (IPFM) estimates a HEU consumption of 85 kg per year for the same time period. While the method seems valid given that the estimations are of the same order of magnitude, the difference is still considerable. This is not surprising considering the speculative nature of many input parameters, especially the transport and processing time as well as the annual Tc-99m consumption. To assess the influence of these parameters, a sensitivity analysis was performed for the most important ones (see Table 3).

Transport duration [d]		No. of withdrawals		Neutron flux [$s^{-1}cm^{-2}$]		Irradiation time [d]	
value	*HEU [kg/a]*	*value*	*HEU [kg/a]*	*value*	*HEU [kg/a]*	*value*	*HEU [kg/a]*
2	5.6	3	22	5e13	31	2	28
6	15	6	15	1e14	15	5	15
10	42	12	12	2e14	7.7	10	12

Table 3: Influence of different parameter values on the calculated annual HEU demand for Mo-99 production.

This analysis shows how the amount of required HEU according to the selected parameters is close to the realistically possible minimum. It increases significantly if a longer transport time and less Tc-99m withdrawals are assumed. On the contrary, the increase of withdrawals runs into a saturation at the minimum HEU consumption rate of 12 kg/a. The irradiation parameters are less relevant to the result: the amount of required HEU is inversely proportional to the neutron flux, which on the other hand is a parameter of high confidence as it can be confirmed by Saey [7]. Assuming a shorter irradiation time would almost double the HEU demand and should be carefully considered, as reactor irradiation time is very expensive for the Mo-99 producer. On the other hand, longer irradiation times would not significantly decrease the required HEU as the output of Mo-99 is limited by its short half life.

Taking these uncertainties into account, it can still safely be concluded that the production of Mo-99 / Tc-99m alone requires most likely about 15 kg of HEU per year. At least 10 kg/a are required, 50 or even 100 kg/a might be used. This HEU, after some cooling time following irradiation, is handled in an environment with a low radiation barrier, and is mostly not recycled, but kept on site in liquid storage. This storage and handling of HEU is relatively small compared to the amounts circulating in other sectors, e. g. in research or naval military reactors. But it is an ongoing proliferation risk that could easily be avoided by converting to a LEU production procedure. If medical isotope consumption continues to increase with 4.5% per year as in the past decade while other stocks of HEU are reduced, the relative importance of medical HEU for nuclear non-proliferation will become more and more relevant.

5. Acknowledgment

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SESSION 6

ENVIRONMENTAL SAMPLING

NUSIMEP-6: Uranium isotope amount ratios in uranium particles

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Abstract:

The IRMM Nuclear Signatures Interlaboratory Measurement Evaluation Programme (NUSIMEP) is an external quality control programme organised by IRMM with the objective of providing materials for measurements of trace amounts of nuclear materials in environmental matrices. Measurements of the isotopic ratios of the elements uranium and plutonium in small amounts, such as typically found in environmental samples, are required for nuclear safeguards, for the control of environmental contamination and for the detection of nuclear proliferation. Participants in NUSIMEP compare their reported measurement results with independent external certified reference values with demonstrated traceability and uncertainty, as evaluated according to international guidelines.
NUSIMEP-6 focused on measurements of Uranium isotope amount ratios in uranium particles aiming to support laboratories involved in uranium particle analysis. It was the first NUSIMEP on particle analysis and particularly also organised for the IAEA network of analytical laboratories for environmental sampling (NWAL) as part of the IRMM activities in the frame of the EC support programme to the IAEA. The NUSIMEP test samples were prepared by controlled hydrolysis of well certified uranium hexafluoride in an aerosol deposition chamber at IRMM. Participating laboratories in NUSIMEP-6 received a test sample of uranium particles on a graphite planchet with undisclosed isotope amount ratio values $n(^{234}U)/n(^{238}U)$, $n(^{235}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$. The uranium isotope amount ratios were to be measured using their routine analytical procedures. Measurement of the major ratio $n(^{235}U)/n(^{238}U)$ was obligatory; measurements of the minor ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ were optional. 15 institutes reported measurement results, among those 7 NWAL laboratories. The analytical methods applied were SIMS, (FT)-TIMS, LA-ICP-MS and alpha spectrometry. The participants' measurement results were evaluated against the certified reference values. The results of NUSIMEP-6 confirm the capability of laboratories in measuring the major ratio $n(^{235}U)/n(^{238}U)$ in uranium particles. More difficulties were observed for the minor isotope ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$. In addition feedback from the participants was collected in view of improvements and optimisation of future NUSIMEP interlaboratory comparisons for uranium isotope amount ratios in uranium particles.

Keywords: Interlaboratory Comparisons; environmental sampling; isotope ratio measurements; nuclear safeguards; quality control

1. Introduction

Nuclear safeguards arrangements exist on international level under the protocols of the International Atomic Energy Agency (IAEA) [1] on European Union level under the Euratom Treaty [2] and on regional levels. The INFCIRC/540 [3], also referred to as the Additional Protocol (AP) moved the focus from exclusively accounting for known quantities of fissile material towards a more qualitative system that is able to provide a comprehensive picture of a state's nuclear activities. As part of the Additional Protocol, environmental sampling has become an important tool for the detection of non-declared nuclear activities. One extensively developed technique in environmental sampling (ES) makes use of pieces of cotton cloth called swipes to wipe surfaces inside and around a nuclear facility. The dust

collected on these swipes typically contains micrometer-sized uranium particles with an isotopic composition characteristic for the processes at the inspected facility. Measurements of minor isotope abundance ratios of uranium in those particles, may provide additional information about equipment or plant design, indicate information about irradiation history, and also help to evaluate mixing and decay scenarios. Major and minor uranium isotope ratios in environmental samples collected by inspectors are measured by the IAEA's Seibersdorf Analytical Laboratory (SAL) in Austria and the Network of Analytical Laboratories (NWAL) [4].

Recently a workshop organised by the ESARDA Working Group on Standards and Techniques for Destructive Analysis (WG DA) was held at IRMM on measurements of minor isotopes in uranium bulk and particle samples [5]. Participants in this workshop came from the main European and international nuclear safeguards organisations, nuclear measurement laboratories as well as from geochemistry and environmental sciences institutes. During this workshop it was stressed that considering the potential consequences of particle analyses in nuclear safeguards, bio- and earth sciences, these measurements need to be subjected to a rigorous quality management system. The reliability and comparability of measurement results of isotope ratios in uranium particles need to be guaranteed and monitored via the correct use of reference materials and quality tools. Currently it is clearly a significant drawback for laboratories involved in particle analysis that such materials are not available. Therefore special attention has been given recently at IRMM to the development of uranium particle reference materials and quality control samples for the analysis for environmental samples [6, 7].
To address the needs from international safeguards authorities and research institutions IRMM organised the first NUSIMEP interlaboratory comparison on isotope ratio measurements in uranium particles.

2. NUSIMEP

The IRMM Nuclear Signatures Interlaboratory Measurement Evaluation Programme (NUSIMEP) is an external quality control programme organised by the Joint Research Centre - Institute for Reference Materials and Measurements (IRMM) to support the growing need to trace and measure the isotopic abundances of elements characteristic for the nuclear fuel cycle present in trace amounts in the environment. Measurements of the isotopic ratios of the elements uranium and plutonium in small amounts, such as typically found in environmental samples, are required for nuclear safeguards, for the control of environmental contamination and for the detection of nuclear proliferation.
Laboratories analysing environmental samples are invited to participate in these external NUSIMEP quality control exercises to demonstrate and assess their ability to carry out precise measurements in particular on trace amounts of uranium and plutonium. Through this and similar programmes, the degree of equivalence of measurements of individual laboratories can be ascertained. IRMM is an interlaboratory comparison organiser accredited according to ISO Guide 43-1 [8]. Reports on NUSIMEP interlaboratory comparisons can be found on the IRMM web-site [9].

3. Scope and aim

Measurements of the isotopic ratios of the elements uranium and plutonium in small amounts, such as typically found in environmental samples, are required for nuclear safeguards, for the control of environmental contamination and for the detection of nuclear proliferation. NUSIMEP-6 aims at laboratories carrying out particle analysis in these various application fields. Particular emphasis was given to participation of the IAEA network of analytical laboratories for environmental sampling (NWAL) in support to nuclear safeguards arrangements. Participation of the NWAL laboratories in this NUSIMEP interlaboratory comparison was formally recommended by the IAEA at the IAEA Technical Meeting on Particle Analysis of Environmental Samples for Safeguards. NUSIMEP-6 is a pilot interlaboratory comparison that not only should picture the measurement capabilities of the participating laboratories at a certain point in time, but also collect feedback from the participants towards future improvements and needs, which made this pilot interlaboratory comparison a very useful exercise for the coordinators as well as for the participating laboratories. Measurands in NUSIMEP-6 is the isotope amount ratio values $n(^{234}U)/n(^{238}U)$, $n(^{235}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$. The matrix is uranium particles on a graphite planchet.

4. Test material

4.1. General remarks

The process applied at IRMM to produce uranium particles from well certified uranium hexafluoride (UF_6) is described in detail in [6]. In the meantime an improved aerosol deposition chamber was developed at IRMM to control the relative humidity and temperature during the production of uranium particles from the controlled hydrolysis of uranium hexafluoride (UF_6) aiming at the production of single uranium particles in the 1µm range. This new aerosol deposition chamber was used to produce the reference particles for the NUSIMEP-6 interlaboratory comparison.

4.2. Preparation

A depleted UF_6 reference material with a $n(^{235}U)/n(^{238}U)$ ratio of 0.0070439(35), stored in a monel (cupper-nickel alloy) ampoule, was used for NUSIMEP-6. Milligram amounts of this UF_6 reference material was distilled into a glass vial.

After transfer, the glass vial containing the gaseous UF_6 was placed into an aerosol deposition chamber. The apparatus consists of an aluminium cylindrical reaction chamber with lids in Plexiglas (Fig. 1). The glass vial containing the UF_6 reference material was broken by a pin. In this way, the UF_6 was released and subsequently hydrolyzed.

Fig. 1: Set-up of the aerosol deposition chamber

The humidity and the temperature of the air inside the chamber were monitored by a hygrometer (Rotronic). The reaction between the released uranium hexafluoride and the atmospheric moisture in the deposition chamber proceeds very rapidly to form solid uranium oxyfluoride particles and hydrogen fluoride. The simplified overall equation is as follows:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

At the base of the aerosol deposition chamber, a retractable platform containing 6 graphite discs of 25 mm diameter was used to collect the settling uranium oxyfluoride particles. The particle morphology was then verified by scanning electron microscopy (SEM) for all of the NUSIMEP-6 samples.

The NUSIMEP-6 samples were put in boxes with silica-gel and sealed in plastic bags and stored at room temperature until dispatch.

4.3. Verification, homogeneity and stability

The NUSIMEP-6 uranium particles are produced from a well certified uranium hexafluoride reference material. This reference material was certified in the chemical form of uranium hexafluoride by gas mass spectrometry for the $n(^{235}U)/n(^{238}U)$ ratio and in the form of uranium nitrate by thermal ionisation

mass spectrometry for the $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ ratios. From previous studies it was known that no isotopic effects occur during aerosol deposition of uranium hexafluoride [6]. Nevertheless, measurements on blank planchets and on samples taken from each badge produced were performed using thermal ionisation mass spectrometry (TIMS). A homogeneity and stability study was carried out according to relevant ISO guidelines [10]. As a result of these studies the uranium test material was found to be a suitable test material for NUSIMEP-6 [11].

5. Participant invitation, registration and information

Participation of the NWAL laboratories in this NUSIMEP interlaboratory comparison was formally recommended by the IAEA at the IAEA Technical Meeting on Particle Analysis of Environmental Samples for Safeguards. Furthermore NUSIMEP-6 was announced in relevant conferences and meetings and published on the IRMM website. Measurement of the major ratio $n(^{235}U)/n(^{238}U)$ was obligatory measurements of the minor ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ were optional. Participants were invited to follow their routine procedures. 20 participants representing 15 institutes from all recognised nuclear weapon states participated in NUSIMEP-6. Among those participants were 7 NWAL's. The mission of half of the laboratories participating in NUSIMEP-6 is to carry out measurements for fissile material control or safeguards but also for environmental sciences. The other participants are from the fields of occupational health, research & development, geosciences, material analysis. All but 2 participants indicated that their laboratories are either accredited and/or authorised for this type of measurements. This suggests that NUSIMEP-6 is a useful and representative study for the current capability of laboratories in the field of uranium particle analysis.

7. Reported results

7.2. Measurement results

Fig. 2 – 4 display the results from the NWAL laboratories.
The graph for $n(^{234}U)/n(^{238}U)$ shows a roughly normal distribution with no irregularities. Most of the NWALs stayed within a 5% deviation from the certified value. The graph for $n(^{235}U)/n(^{238}U)$ also shows roughly a normal distribution with no irregularities. Most of the NWALs even stayed within a 1% deviation. In case of $n(^{236}U)/n(^{238}U)$ 2 of the NWALs reported results within +/- 100% deviation from the reference value. The majority of the participants reported ratios that were too large or reported an upper limit for $n(^{236}U)/n(^{238}U)$. According to safeguards requirements NWAL laboratories can report an upper limit in case the isotope amount fraction of the minor isotopes is below 1ppm, which is the case in NUSIMEP-6; $n(^{236}U)/n(U) < 1ppm$.

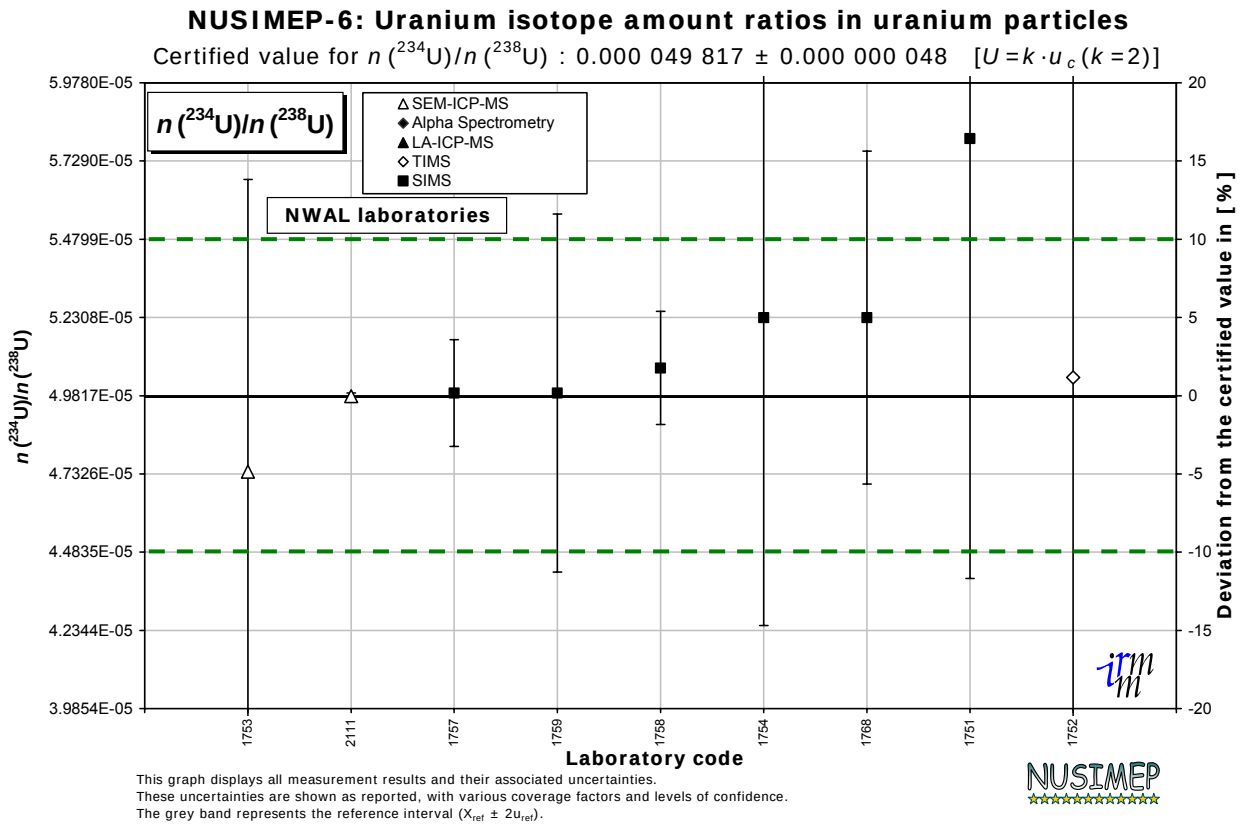

Fig. 2: Graph of reported $n(^{234}U)/n(^{238}U)$ by NWAL laboratories

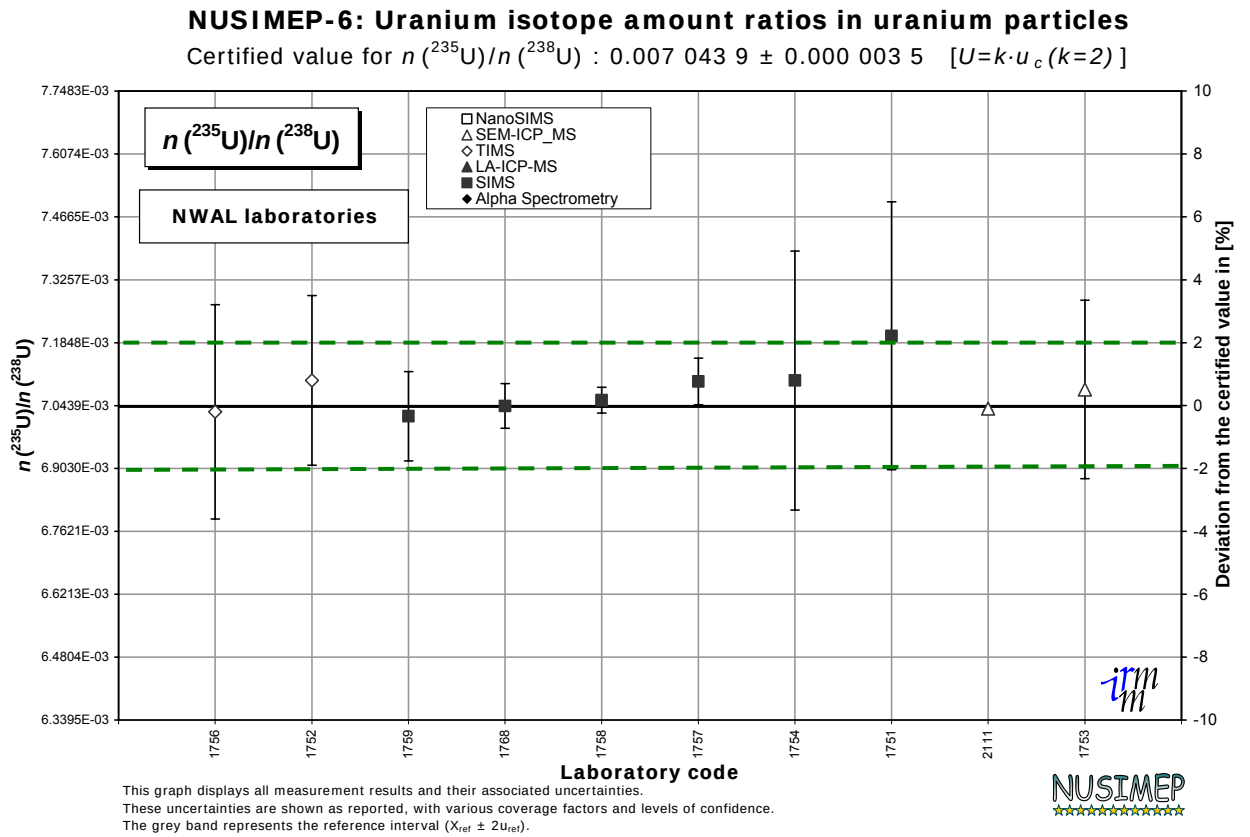

Fig. 3: Graph of reported $n(^{235}U)/n(^{238}U)$ by NWAL laboratories

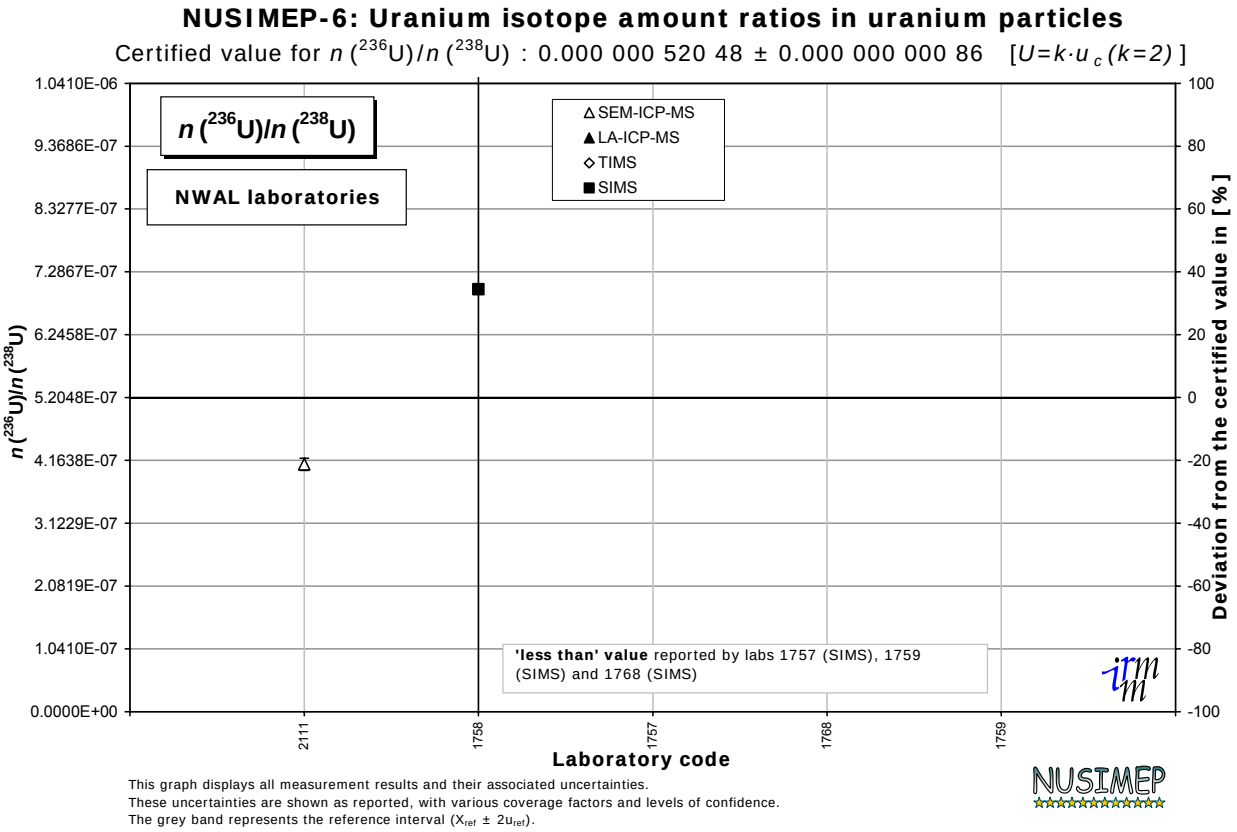

Fig. 4: Graph of reported $n(^{236}U)/n(^{238}U)$ by NWAL laboratories

8. Scoring of results

8.1. The scores and their settings

Individual laboratory performance is expressed in terms of z and zeta scores in accordance with ISO 13528 [10]:

$$z = \frac{x_{lab} - X_{ref}}{\hat{\sigma}} \qquad \text{and} \qquad zeta = \frac{x_{lab} - X_{ref}}{\sqrt{u_{ref}^2 + u_{lab}^2}}$$

Where

x_{lab} is the measurement result reported by a participant
X_{ref} is the certified reference value (assigned value)
u_{ref} is the standard uncertainty of the reference value
u_{lab} is the standard uncertainty reported by a participant
$\hat{\sigma}$ is the standard deviation for proficiency assessment

Both scores can be interpreted as: satisfactory result for |score|≤2, questionable result for 2<|score|≤3 and unsatisfactory result for |score|>3.

The further interpretation of the z score and zeta score is explained in the NUSIMEP-6 report [11].

8.2. Scoring the reported measurement results

A z score was calculated for all participants except for those who reported no value or an upper limit, "<" value. A zeta score was calculated for results that were accompanied by an uncertainty statement.

Table 1 summarises the scores per isotope amount ratio.

A large share of participants reported satisfactory measurement results for the $n(^{235}U)/n(^{238}U)$ and $n(^{234}U)/n(^{238}U)$ isotope amount ratios, and only a small share unsatisfactory results. It can be concluded that the participants performed quite well in NUSIMEP-6. For the small $n(^{236}U)/n(^{238}U)$ isotope amount ratio only 2 participants had satisfactory z and zeta scores.

	z score				zeta score				both z and zeta scores
	S	Q	U	n (*)	S	Q	U	n (*)	S
$n(^{234}U)/n(^{238}U)$	86%	-	14%	14	86%	-	14%	14	12
$n(^{235}U)/n(^{238}U)$	74%	16%	10%	19	80%	5%	15%	19	13
$n(^{236}U)/n(^{238}U)$	67%	-	33%	6	50%	-	50%	6	2

(*) n is the number of results for which a score was given.
The total number of participants (with and without a score) is 20.

Table 1: Overview of scores: S(atisfactory), Q(uestionable), U(nsatisfactory)

9. Further information extracted from the results

In addition to submission of the results, the participants were asked to answer a number of questions relating to the measurements. All participants completed the questionnaire and contained useful information concerning methods of analysis, correction for mass fractionation / mass bias, analytical procedure, quality system and the use of standards, determination of uncertainty, future NUSIMEP ILCs on particles. The preferred instrumental technique was secondary ion mass spectrometry (SIMS), follwed by ICP-MS and TIMS. Participants' replys to this questionnaire are summarised in the NUSIMEP-6 report [11].

10. Feedback

Since NUSIMEP-6 was a pilot intercomparison, the first coordinated by IRMM of its kind, the participants were particularly encouraged to provide feedback to the ILC coordinators. 2 participants would have preferred that the planchet was fixed in the plastic box. Most of the participants also mentioned that particles in the sample were less than 1µm, with only very few up to 5µm, which made particle analysis difficult, particularly for the minor isotopes. It was also suggested to use monodispersed particles for SIMS analysis. The planchets were too heavily loaded for FT-TIMS measurements and too small. Some participants requested a more explicit measurement protocol from the ILC organisers, including the possibility to report results for individual particles. Furthermore streaks and smears of uranium were present on some planchets. This likely occurred during aerosol deposition. One participant also noted the presence of an isobaric at mass 237 that affected their ability to derive meaningful results for $n(^{236}U)/n(^{238}U)$.

11. Conclusion

There is an increased need for information in order to verify not only the amounts of nuclear material, but also the consistency of information as provided by states or plant operators. To this end, techniques like particle analysis have been implemented. The fundamental importance of measurements of major and minor uranium isotopes in environmental sampling (ES) was recently also stressed by the IAEA during the Workshop on Measurements of Minor Isotopes in Uranium Organized by the ESARDA Working Group on Standards and Techniques for Destructive Analysis (WG DA), since minor uranium ratios are measured in almost all of the environmental samples [5]. The major and minor isotope amount ratios in uranium were the measurands under investigation in NUSIMEP-6. The measurement

capabilities in uranium particles for $n(^{235}U)/n(^{238}U)$ and $n(^{234}U)/n(^{238}U)$ were very good, particularly for the NWAL laboratories. Only a few results were reported for the small $n(^{236}U)/n(^{238}U)$ isotope amount ratio. Differences are observed in the uncertainty estimates provided by the participants, even when using same instrumental techniques. At present there are no official safeguard requirements on the uncertainties of measurements in particle analysis. The concept of “Target Values for Uncertainty Components” for element and isotope assay of nuclear materials was originally conceived in 1979 by the Working Group on Techniques and Standards for Destructive Analysis (WGDA) of the European Safeguards Research and Development Association (ESARDA) and matured gradually during many years [12]. The definition of performance standards for measurements in particle analysis appears highly recommended. The ESARDA WG DA undertakes establishing such performance standards as guidance for measurement laboratories [13].

The aim of the first Nuclear Signatures Interlaboratory Measurement Evaluation Programme was to study the capability of analytical laboratories to measure uranium isotope amount ratios in uranium particles. NUSIMEP-6 was intended as a pilot study on uranium particle analysis. One objective of NUSIMEP-6 was also to collect feedback from the participants in view of optimisation of uranium reference particle production. From the feedback it became clear that the parameters for uranium particles produced via aerosol deposition of well characterised UF_6 standards still needs some improvement in view of controlling the particle density, particle size and avoiding additional uranium films on the planchets. The needs expressed by the NUSIMEP-6 participants for reference particles in future NUSIMEP ILCs are manifold. For safeguards analysis a sample with uranium of different enrichment and/or uranium mixed with a few plutonium particles would be required. The analysis of two different samples, one with monodispersed particles in combination with “close to real-life” particles produced via aerosol deposition would be beneficial for NWAL laboratories using SIMS, furthermore also the analysis of uranium oxyfluoride particles. The feedback of the NUSIMEP-6 participants was very much appreciated by the ILC coordinators in view of the next NUSIMEP on particle analysis, to be launched end of 2009. Optimisation in particle production and characterisation are already ongoing at IRMM. Furthermore, IRMM established close cooperation with ITU and the IAEA in defining the needs for well characterised uranium particle test samples and the feasibility of production and certification.

The Safeguards Analytical Laboratory (SAL) particularly acknowledged that NUSIMEP-6 was a very good exercise for SAL, the NWAL laboratories and other experts in this field. The safeguards analytical laboratory faces more and more challenges in the field of environmental sampling and is in need of reference particles of both uranium and plutonium for instrument calibration, quality control and interlaboratory comparisons with an isotopic composition representative for the range of particles found in nuclear installations.

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Sample preparation method development for analysis of safeguards swipe samples by inductively coupled plasma mass spectrometry

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Abstract:

International safeguards have been applied for over 30 years to control nuclear activities. Following the recon of undeclared nuclear activities in Iraq in 1991, strengthening of the safeguards system became necessary (1995). From 1996 the detection of illegal nuclear activities using sensitive and precise analysis of swipe samples became one of the parts of IAEA inspections.
The aim of this study was the development of a fast and easy sample preparation and analysis method for the bulk analysis of swipe samples. For the sample preparation microwave-assisted digestion followed by extraction chromatography with TRU® resin was applied. The analytes and the isotopic composition were determined by inductively coupled plasma sector field mass spectrometry (ICP-SFMS).
The analytical performance of the method is in good agreement with the requirements (accuracy, precision, repeatability) of IAEA-NWAL and can be applied for routine analysis. The low detection limits achieved enable the analysis of the isotope ratios and isotope concentrations of U and Pu isotopes (^{234}U, ^{235}U, ^{236}U, ^{238}U and ^{239}Pu, ^{240}Pu, ^{241}Pu) and other radionuclides (e.g. Th, Am, Np) present in ultratrace amount in swipe samples. The method was tested with real swipe samples taken at Hungarian nuclear facilities.

Keywords: safeguards; swipe samples; bulk analysis; ICP-SFMS

1. Introduction

International safeguards have been applied about 30 years in order to verify that nuclear materials declared by a State to the IAEA are used for peaceful purposes only [1].
In 1991 IAEA and inspectors were combing the rubble of Iraq's nuclear installations looking for evidence of a secret program to produce atomic bombs, something expressly forbidden by Iraq's ratification of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT). This case opened a new chapter in the development of international nuclear safeguards. States in 1995 adopted measures for a strengthened safeguards system that authorize and equip inspectors to assure that any undeclared nuclear activities would not be overlooked [2].
One of the principal strengthening measures studied under this program was the use of environmental sampling and analysis to detect nuclear signatures which might reveal undeclared activities [1].
Each nuclear activity leaves a tell-tale fingerprint on the environment that can be picked up by taking environmental samples. The most effective sampling type is the swipe samples. The Agency regularly takes such samples in the field and sends them to laboratories for detection of any traces of nuclear material. [3]. A big amount of information can be obtained from the small amount of material collected in one sample that inspectors swipe on a 10 x 10 cm piece of cotton cloth [2].
Bulk analysis of swipe samples gives information about the average concentration or isotopic composition of the whole sample. For this purpose, some instrumental methods: based on radiometry or mass spectrometry, e.g.: thermal ionization mass spectrometry (TIMS), secondary ion mass spectrometry (SIMS), accelerator mass spectrometry (AMS) has been used for the identification of isotopes in swipe samples. However, several of these methods require very expensive instrumentation, difficult operation and time consuming and tedious sample preparation [4].

Inductively coupled plasma mass spectrometry (ICP-MS) is at present the most frequently used inorganic mass spectrometric technique for concentration and isotope ratio measurements down to fg g^{-1} level. This powerful analytical technique is also increasingly used for the measurement of long-lived radionuclides, including plutonium [4], [5].
The aim of this study was to develop a relatively fast and easy sample preparation and analysis method for bulk analysis of environmental swipe samples using inductively coupled plasma sector field mass spectrometry (ICP-SFMS). The method was developed for analysis of uranium and plutonium isotopes, and it was tested with real swipe samples taken at Hungarian nuclear facilities.

2. Experimental

2.1. Reagents, standard solutions, tracers and samples

The ^{242}Pu (NIST 4334F, USA), ^{243}Am (Amersham International, UK), ^{233}U (New Brunswick Laboratory, USA) and ^{232}Th (SPEX CertiPrep, USA) isotopic standards were used to spike the samples. Multielement standard solution (Merck, Germany) was used for the optimization of the ICP-SFMS instrument. Natural uranium solution was used to correct for mass discrimination. The TRU (100–150 µm particle size, active component: octylphenyl-N,N-di-isobutyl carbamoylphosphine oxide dissolved in tri-n-butyl phosphate) extraction chromatographic resin were supplied by Eichrom Technologies Inc. (USA). For the analysis, 0.5 ml of the resin was placed in plastic Pasteur pipette (diameter: 4 mm, length: 12 mm) and plugged with a piece of cotton cloth on base and top of the resin. All reagents used were of analytical grade. Nitric, hydrochloric, hydrofluoric, perchloric acids and hydrogen peroxide used in the final solution for ICP-MS analysis were Suprapur grade (Merck, Germany). Solid salts; oxalic acid and ascorbic acid originated from Merck (Germany), too. For dilution of reagents and samples de-ionised water was used (Milli-Q System, Millipore, USA). All sample preparation and measurement procedures were carried out under clean room conditions (class 100000).

2.2. Apparatus

Digestion of the samples was carried out applying a MARS5 microwave digestion system (CEM Corp., USA). Plutonium isotope ratios and concentrations were determined using a double focusing magnetic sector inductively coupled plasma mass spectrometer equipped with a single electron multiplier (ELEMENT2, Thermo Electron Corp., Germany). All measurements were carried out in low resolution mode ($m/\Delta m$= 300) using a low-flow T1-H nebulizer in self aspirating mode (flow rate approximately 60 µ l min^{-1}) in combination with a desolvation unit (Aridus, CETAC Technologies Inc., USA) that removes most of the solvent thus significantly decreases hydride and oxide interferences. Optimized operating parameters and data acquisition parameters are summarized in Table 1.

Instrument parameters	
RF power (W)	1340
Cooling gas flow rate (l min^{-1})	15.4
Auxiliary gas flow rate (l min^{-1})	1.01
Nebulizer gas flow rate (l min^{-1})	0.965
Sample introduction (Aridus system with T1-H nebulizer)	
Solution uptake rate (µl min^{-1})	60
Sweep gas flow rate (l min^{-1})	5.42
Spray chamber temperature (°C)	80
Membrane temperature (°C)	160
Data collection	
Resolution	300
Runs and passes	5 x 5
Mass window (%)	5
Samples per peak	200
Search window (%)	60
Integration window (%)	5
Integration type	Average
Scan type	E-Scan

Table 1: Optimized operating condition and data acquisition parameters of ICP-SFMS analysis

2.3. Procedure

The total swipe sample was weighted into a PFA microwave digestion vessel (1.8 g) and known amount of ^{242}Pu, ^{243}Am, ^{233}U, ^{232}Th tracers for isotopic dilution quantification were added to the samples by weight. Following the addition of 10 ml HNO_3, samples were predigested for a night. After that samples were digested using MARS5 microwave digestion system. The samples were heated in three step: 1. up to 80 °C in 15 minutes and held at that temperature for 2 minutes; 2. up to 120 °C in 20 minutes and held at that temperature for 10 minutes; 3. up to 150 °C in 15 minutes and held at that temperature for 10 minutes. The applied microwave power was 300 and 600 W, respectively. The lower energy and longer heating time was necessary because of the large amount of the samples (~ 2 g).

Sample were washed with Milli-Q water into 50 ml PP tubes and diluted to approximately 40 ml. After the dilution 0.5 ml of 25 w/w% $NaNO_2$-solution was added to the samples, which stabilizes Pu in the Pu(IV) state within 10 minutes. The plutonium, uranium and the other radionuclides were separated by extraction chromatography using TRU resin. After conditioning of the column with 5 ml of 3M HNO_3 the sample was loaded on the column. After loading the tube and the column were rinsed twice with 2.5 ml of 3M HNO_3. Firstly, americium was eluted by 5 ml 4M HCl. After that plutonium was eluted by 15 ml of 0.1M ascorbic acid/4M HCl. The next fraction was contained thorium and neptunium. Elution was achieved with 8 ml of 0.2M HF/4M HCl. Finally, uranium was eluted by 15 ml of 0.05M oxalic acid/0.1M HCl. Fractions were collected in PFA beakers and were repeatedly evaporated to almost dryness (3 times) with successive addition of 2 ml of ultrapure HNO_3, 1 ml of H_2O_2 and some droplets of perchloric acid to remove HF and organic contents. The residue was dissolved in dilute ultrapure nitric acid (1-2 ml) with gentle heating on a hotplate. The schematic diagram of the sample preparation is shown in Fig. 1.

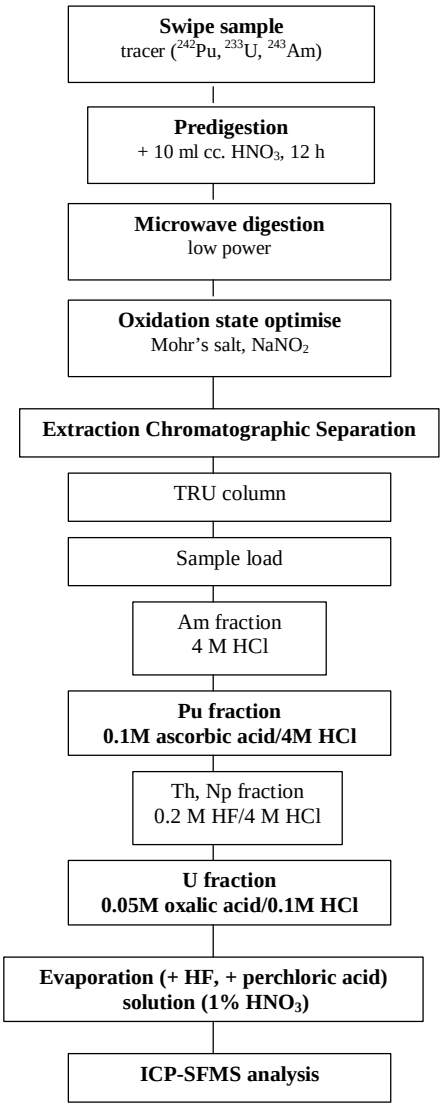

Figure1: Schematics of the sample preparation

3. Results and discussion

Sample preparation method of swipe samples is usually a tedious procedure. The commonly used first step is the ashing procedure which is relatively time consuming. The aim of this study was to decrease the numbers and time of the sample preparation steps. Another aim was to separate and pre-concentration of the radionuclides.

Sample preparation time can be redduced using only microwave digestion procedure. Using this method with lower energy the ashing step is not necessary and digestion of the whole swipe sample (without cutting) is possible.

Extraction chromatography is a technique that is ideally suited for the separation of radionuclides also in swipe samples. This technique combines the selectivity of liquid-liquid extraction with the ease of operation of column chromatography. It is quick and easy to use and it has low chemical needs which provide low background for the instrumental measurements.

During the development of the extraction chromatographic separation method we tested several reducers (oxalic acid, ascorbic acid, NH_4I, $NH_2OH.HCl$) for elution of plutonium and uranium. Finally, ascorbic acid for plutonium and oxalic acid for uranium can be used in acidic medium. By the method development the most important aim was to separate the uranium and plutonium. Application of the

chromatographic separation further purifies the plutonium fraction removing the leftover Th, U and other components (matrix) that could possibly form molecular interference (e.g., $^{238}U^{1}H^{+}$, $^{207}Pb^{16}O_{2}^{+}$, etc.) in mass spectrometric analysis.
Figure 2 shows the optimized elution profile. It can be seen that application of the optimized extraction chromatographic method separation of the radionuclides: especially the plutonium and uranium is fully possible.

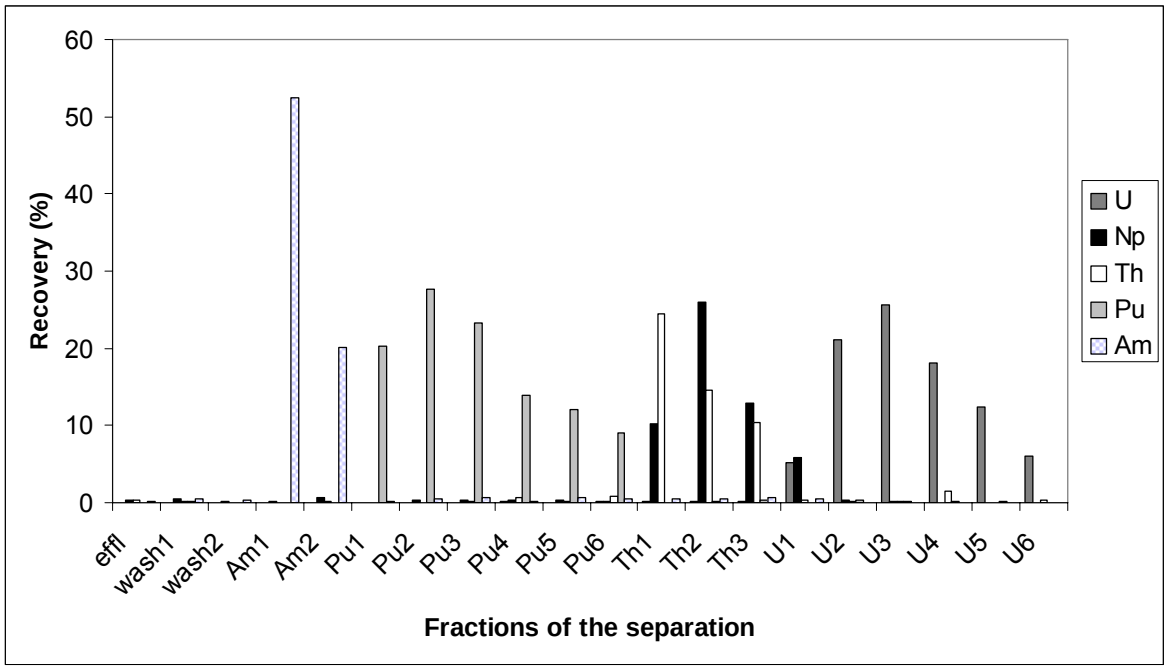

Figure 2: Optimized elution profile of the separated radionuclides

Analytical performance parameters of the method are adequate and in good agreement with the requirements of IAEA. The limits of detection (LOD) of the ICP-SFMS analysis are calculated for 1 gram of sample for easier comparison (Table 2). Calculation of LOD was based on three times the standard deviation of the method blank.

Isotope	Detection limit	IAEA/NWAL requirements
^{239}**Pu**	7.5 fg	Pu ≤ 10 fg/ swipe
^{240}**Pu**	2.4 fg	
^{241}**Pu**	1.1 fg	
^{238}**U**	1 ng	U ≤ 0.1-5 ng/ swipe
^{235}**U**	11.4 pg	
^{234}**U**	0.14 pg	
^{236}**U**	0.083 pg	

Table 2: The obtained detection limits of the method and the IAEA requirements

The precision of the isotope ratios by plutonium varied between 7–10% and by uranium varied between 0.23-3% (at 95% confidence level) in the case of $^{240}Pu/^{239}Pu$, $^{235}U/^{238}U$, $^{234}U/^{238}U$ and $^{236}U/^{238}U$ ratios, respectively. The IAEA-NWAL requirements are the following: ≤ 1 and 10% by uranium isotope ratios and ≤ 10% by plutonium isotope ratios.
The accuracy of the isotope ratios was validated by the analysis of synthetic saline solution with certified isotopic composition. Samples were originated from the National Nuclear Signatures Inter-laboratory Measurement Evaluation Programme (NUSIMEP, IRMM, Belgium). The determined isotope

ratios ($^{235}U/^{238}U$, $^{240}Pu/^{239}Pu$) in the case of uranium and plutonium are presented in Figure 3 and Figure 4. The recommended values are also indicated. The measured results are in good agreement with the reference values, however, the $^{240}Pu/^{239}Pu$ results by the sample of N5-B have relatively high difference and uncertainty because of the low signal intensity.

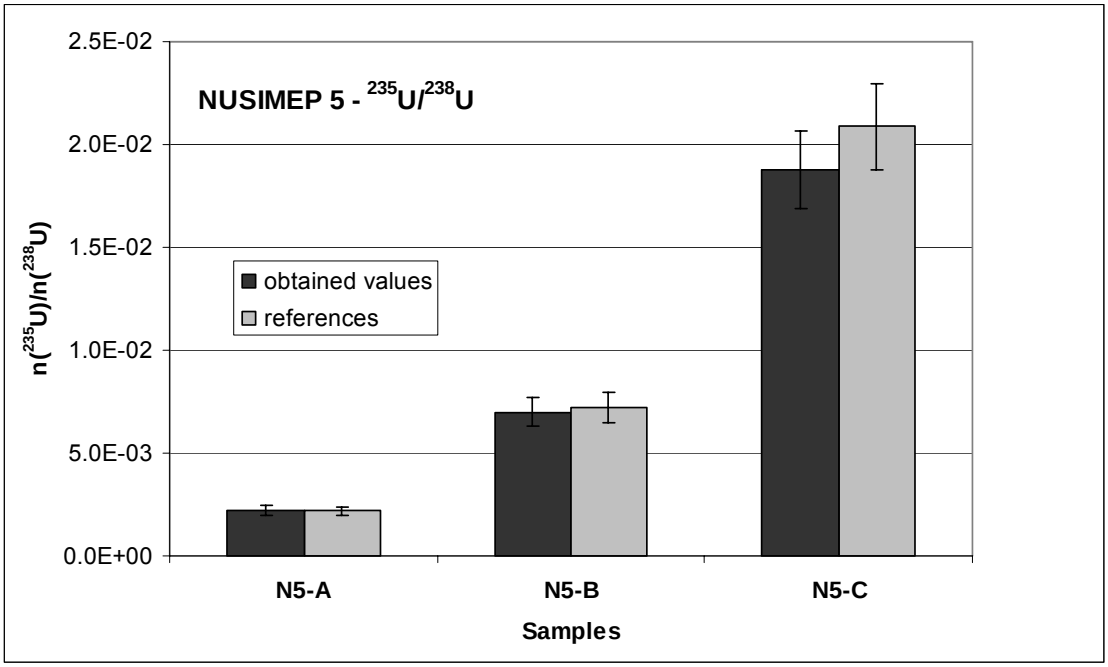

Figure 3: Obtained and reference isotope ratio values by the samples of NUSIMEP-5 (N5-A, N5-B and N5-C) in the case of uranium isotope ratios

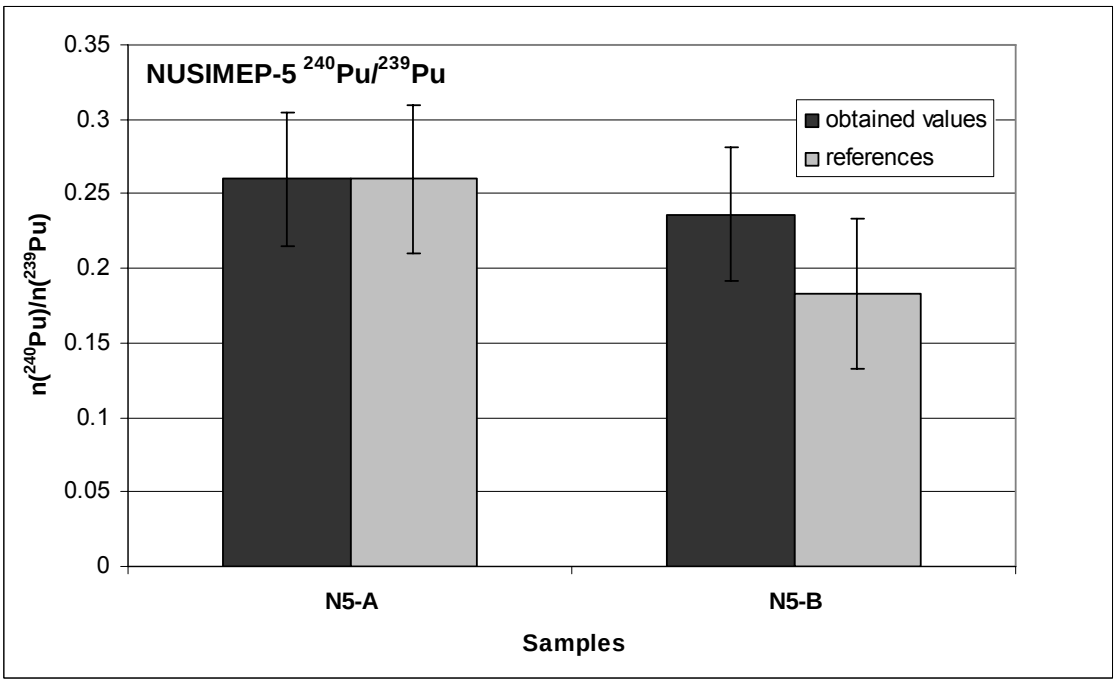

Figure 3: Obtained and reference isotope ratio values by the samples of NUSIMEP-5 (N5-A, N5-B and N5-C) in the case of plutonium isotope ratios

4. Conclusions

A rapid and simple sample preparation method has been developed for the determination of plutonium, uranium and other radionuclides in environmental swipe samples by ICP-SFMS.
Shorter sample preparation using only microwave digestion and an easy and fast extraction chromatography step were used for separation and preconcentration of radionuclides in the samples.
The analytical performance of the method agrees with the requirements (accuracy, precision, repeatability) of IAEA-NWAL and can be applied for routine analysis. The low detection limits achieved enable the analysis of the isotope ratios and isotope concentrations of U and Pu isotopes and other radionuclides. The method was tested with real swipe samples taken at Hungarian nuclear facilities.

5. Acknowledgements

This study was supported by the Hungarian Atomic Energy Authority. Zsolt Stefánka thanks the Bolyai János Scholarship from the Hungarian Academy of Sciences for its support, Tamás Bíró (Institute of Isotopes) is thanked for the helpful discussion and the proposal of the work.

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Isotope Ratios Measurements in Environmental Samples using Thermal Ionization Mass Spectrometry (TIMS) and Filament Carburization

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Abstract:

Thermal Ionization Mass Spectrometry (TIMS) is a widely used mass spectrometric technique for the determination of the isotopic composition of plutonium. Moreover, advances in the last decades have led to increasingly high precision and accuracy of the isotope ratio measurements using multiple ion counting detection system. High precision plutonium isotope measurements would allow distinguishing different types of plutonium contamination and could be used as a finger print to track various sources of plutonium in the environment. To measure plutonium isotope ratios with high precision and accuracy, stable ion beams with high intensity are desired. This is sometimes difficult to achieve, especially when measuring plutonium in low amounts.
In this study, carburization of the Re filaments was used to increase the ionization efficiency of plutonium. Carburized filaments were prepared in a special vacuum chamber in which benzene vapour was introduced as a pure source of a carbon. As a result of the carburization of the filaments higher efficiencies were obtained. This carburization technique was combined with a multi-dynamic measurement technique using the multiple ion counting system and applied to two sediment samples IAEA 135 and IAEA 368; respectively. The results in term of plutonium isotope ratios are presented and compared with the literature values.

Keywords: TIMS; Pu isotope ratios; carburization; environmental samples

1. Introduction

Various man-made radioactive elements are present in the environment as a result of human activities. Such element is also plutonium which can be regarded as one of the most important nuclides for safeguards. Plutonium is found in the environment as a result of nuclear weapon tests, nuclear reactor accidents, discharges from reprocessing plants, dumping of nuclear waste and accidents with nuclear devices. On the other hand, plutonium is also produced from uranium in nuclear reactors. For assessing different sources of plutonium contamination, information on the isotopic composition is necessary. It is known that the $^{240}Pu/^{239}Pu$ isotope ratio is a good indicator to identify different types of plutonium contamination, however also other plutonium isotope ratios, such as $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$, could be useful as additional information on the production processes of nuclear material. For global fallout the average $^{240}Pu/^{239}Pu$ and $^{241}Pu/^{239}Pu$ isotope ratios are 0.176 ± 0.014 and 0.0086 ± 0.0017, respectively [1]. Weapons grade plutonium can be characterized by $^{240}Pu/^{239}Pu$ ratios < 0.07, whereas in plutonium from nuclear reactors, ratios of 0.4 or higher can be expected [2-5].

For determination of plutonium isotope ratios mass spectrometric techniques such as thermal ionization mass spectrometry (TIMS), accelerator mass spectrometry (AMS) and inductively coupled plasma mass spectrometry (ICP-MS) are most widely used. They offer high precision and accurate

results which would enable to better distinguish between different types of Pu contamination. To measure smallest amounts of plutonium (< 10^{-10} g), high ionization efficiency and a sophisticated detection system are desirable; especially if isotopes having low abundance need to be measured. Thermal ionization mass spectrometry (TIMS) in combination with a multiple ion counting detection (MIC) system holds the potential to meet this requirement.

Different approaches are described in the literature in order to enhance the overall efficiency of plutonium. In this study, carburization of the filaments using pure benzene gas as a carbon provider was tested and implemented at IRMM. Carbon serves as a reducing agent, promoting the production of actinide ions at the expense of undesirable oxide species. The higher proportion of +1 ions resulted in a substantially more intense ion beam for a given amount of material [3, 6]. The carburization technique in combination with the multi-dynamic measurement technique using the MIC system was already applied to the NBL -137 isotopic standards and NUSIMEP 5 inter-laboratory campaign samples [3, 7]. Here, the results for the complete isotopic composition for two reference sediment samples IAEA 135 and IAEA 368 are presented.

2. Experimental

2.1. Chemical separation of Pu

Samples were leached with hot 8M HNO_3 for 3h. After the separation of the leachant and the residue, the actinides were co-precipitated on CaC_2O_4 to remove iron. The oxalate precipitate was fumed several times with conc. HNO_3 and H_2O_2 and dissolved in 2M HNO_3. After the adjustment of the oxidation state of Pu to Pu(IV), the solution was made 8M HNO_3 using conc. HNO_3 and loaded on an anion exchange column. The column was washed with 8M HNO_3 and 9M HCl to remove matrix elements and thorium. Pu(III) was eluted from the column with 0.1M NH_4I-9MHCl and the solution evaporated to near dryness [8]. Pu fraction was further purified with TEVA columns.

2.2. Carburization of the Re filaments

Rhenium filaments were placed into the filament carburization device and evacuated. After a sufficient vacuum pressure was established (< $1 \cdot 10^{-6}$ mbar), the filaments were subjected to a heating routine for degassing and cleaning. After degassing the benzene was introduced into the chamber through a valve and the filament current was adjusted to a 4A, corresponding to a temperature of ca. 1600-1700 ^{o}C. During the carburization, the benzene pressure was kept constant at the pressure of $5 \cdot 10^{-3}$ mbar for 1 hour. After the carburization was complete, the filaments were left to cool overnight before loading Pu solution [3].

Figure 1: Carburization device (left) and a vacuum chamber with a filament magazine (right).

2.3. Triton TIMS

On the IRMM Triton thermal ionization mass spectrometer, nine Faraday cups, one conventional discrete dynode electron multiplier, and seven CDEMs (continuous dynode electron multiplier) were installed. The width of the CDEMs is identical to the standard Faraday cups, and thus the ion counters can be aligned with single unit mass spacing for the measurement of high mass elements such as U and Pu (see Fig. 1). The advantage of multiple ion counting is the simultaneous collection of several isotopes of a given element. It overcomes many of the problems such as transient signal variation in sample emission and ionization, which would significantly reduce the attainable precision of a single collector measurement. For a given sample, a multiple ion counting measurement makes use of a greater number of ions counted for each isotope compared to a peak-jumping measurement using only a single ion counting detector and therefore provides improved counting. Multiple ion counting is thus advantageous for the cases where the sample size is restricted [9, 10].

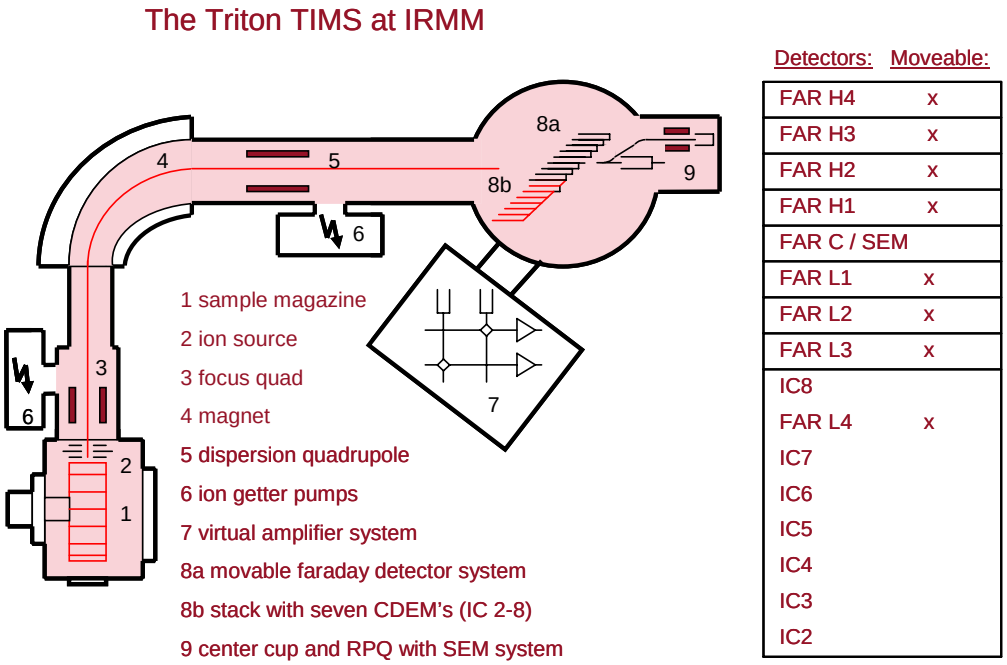

Figure 2: The configuration of the Triton TIMS at IRMM.

2.4. Multi-dynamic measurement technique

Multi-dynamic measurement technique provides improvements in accuracy and precision by applying an internal calibration for the CDEM detectors during the measurement. When all isotopes of interest are detected simultaneously in several steps of a multi-dynamic mass cycle, the isotope ratio can be calculated in a way that the calibration factors of all ion counters are eliminated for so called minor ratios, e.g. for the $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$ ratios in case of plutonium. In contrast, the so called major ratio $^{240}Pu/^{239}Pu$ is measured in a peak-jumping mode using IC3 in steps 1/2, using IC4 in steps 2/3 and using IC5 in steps 3/4, see Table 1, and is therefore also independent on the efficiencies of the CDEMs. A correction factor, the so-called K-factor, for the $^{240}Pu/^{239}Pu$ ratio has to be determined externally using the known $^{240}Pu/^{239}Pu$ isotope ratio of the NBL-137 isotopic standard in order to correct the major ratio for mass fractionation, to be measured using the same technique on the same sample magazine [9, 10]. Table 1 shows the mass cycle for plutonium isotopes in a multi-dynamic measurement.

Channel:	IC2	IC3	IC4	IC5	IC6	IC7
Step:						
1	239	240	241	242		
2		239	240	241	242	
3			239	240	241	242
4				239	240	241

Table 1: The multi-dynamic mass cycle for plutonium measurement

3. Results and Discussion

Different parameters such as carburization current, benzene pressure and time of exposure were studied in order to find the best conditions for an isotopic measurement. The efficiency of plutonium was improved significantly. Moreover, the carburization technique in combination with multi-dynamic measurement technique using the multiple ion counting system (MIC) was successfully applied to NBL-137 isotopic standard and NUSIMEP-5 samples of the inter-laboratory comparison campaign. The detailed results are published in [3].

In this paper, the results of analysis of two reference materials IAEA 135 and IAEA 368 are presented. IAEA 135 (Irish Sea Sediment) is the sediment influenced by the discharges from the reprocessing plant in Sellafield, and IAEA 368 (Pacific Ocean Sediment) is the sediment collected at the French Polynesia, where nuclear tests were carried out.

	IAEA 135 (Irish Sea Sediment)		IAEA 368 (Pacific Ocean Sediment)	
Isotope ratio	Average	Rel. Uc., k=2 [%]	Average	Rel. Uc., k=2 [%]
$^{240}Pu/^{239}Pu$	0.2129	0.23	0.03419	0.34
$^{241}Pu/^{239}Pu$	0.00751	0.36	0.000211	8.3
$^{242}Pu/^{239}Pu$	0.00713	0.51	0.000334	25.3

Table 2: Plutonium isotope ratios in IAEA 135 and IAEA 368.

In table 2, the obtained $^{240}Pu/^{239}Pu$, $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$ isotope ratios are given for IAEA 135 and IAEA 368. The value for the major isotope ratio in IAEA 135, $^{240}Pu/^{239}Pu$, is 0.21290 ± 0.00049. This value is higher than 0.176 ± 0.014, the average ratio due to global fallout [1]. The results agree well with the value of 0.207 ± 0.006 obtained by Lee et al. [11] and 0.211 ± 0.004 given by Muramatsa et al. [5] which were determined using ICP-MS or AMS. The uncertainty of the new TIMS value is about a factor of 10 smaller. The values for $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$ isotope ratios in IAEA 135 were 0.007510 ± 0.000027 and 0.007130 ± 0.00036, respectively. It can also be seen that despite approximately 100 times lower values for minor isotopes in IAEA 135, the uncertainty was still in the same order of magnitude as it was for the major ratio. The $^{242}Pu/^{239}Pu$ agrees within the uncertainty with the value of 0.0068 ± 0.0007 reported by Lee at al. [11], while the $^{241}Pu/^{239}Pu$ isotope ratio was lower than 0.0221 ± 0.0080 observed by Lee.
On the other hand, the $^{240}Pu/^{239}Pu$ value obtained in IAEA 368 was 0.03419 ± 0.00011. This is much lower value than the average for global fallout. In fact the results indicate that the sediment contained weapons grade plutonium ($^{240}Pu/^{239}Pu < 0.07$), which was already assumed. The result for $^{240}Pu/^{239}Pu$ agrees within the uncertainty with the value of 0.043 ± 0.008 reported by Muramatsu et al. [5]. The range of obtained results for $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$ was 10 times lower than in IAEA 135, so higher uncertainties were expected. The values for $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$ isotope ratios in IAEA 368 were 0.000211 ± 0.000018 and 0.000334 ± 0.000085, respectively.

4. Conclusions

It has been shown that the multi-dynamic technique using the multiple on counting system (MIC) and filament carburization could be applied to different environmental samples. As a result of enhanced efficiency of plutonium due to the carburization technique, results with small uncertainties were obtained not only for the $^{240}Pu/^{239}Pu$ isotope ratio but also for $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$. This is very important because the minor ratios provide additional information on the source identification of plutonium in the environment. Unfortunately, little information is available on plutonium isotope ratios in the literature so the results obtained in this study could contribute and complement the "data base" for values for reference materials already existing in the literature.

5. Acknowledgements

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Non-destructive characterization of radioactive particles in safeguard samples

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Abstract:

A measuring system known as PANDA has been designed for non-destructive sample analysis. PANDA introduces new possibilities and techniques for nuclear safeguards. PANDA has two measurement positions in vacuum. The first position includes an electrically cooled high-purity germanium detector (HPGe) and a double-sided silicon strip detector (DSSSD) facing each others. Samples are transported in between the detectors using a linear feedthrough. PANDA's data are recorded in event mode and events are timestamped. Such data acquisition gives flexibility to the analysis. As an example, alpha-gamma coincidence spectra, that are nearly background free due to the underlying physics, can easily be created. In addition, the analysis of the resulting gamma-spectra is simplified since, by definition, it is generated only by the nuclei that decay via alpha-emission.

The DSSSD makes it possible to search efficiently for radioactive particles from the samples. Improved position information makes the isolation of particles more straightforward. As an example, a particle containing 10^{-9} g of ^{239,240}Pu can be detected and located in a few minutes and using the alpha-gamma coincidence technique it is possible to get the first estimate of the isotopic composition in 24 hours.

Keywords: non-destructive analysis, particle analysis, isotope ratios, event mode data acquisition

1. Introduction

STUK - Radiation and Nuclear Safety Authority launched a research programme in 2008 to introduce state-of-the-art spectroscopy tools for sample analysis. In the present article the main focus is in novel techniques for sample measurement. Improved sampling methods are also discussed. Position sensitive Double Sided Silicon Strip Detectors (DSSSD), detectors with ultra thin entrance windows and event-mode data acquisition systems are introduced for sample analysis. The results from the feasibility study of this programme were published in [1]. Improved detection capability of heavy elements was reported. It was also shown that the proposed technique is able to detect and locate a particle containing 10^{-9} g of ^{239,240}Pu in a few minutes. By relying on alpha-gamma coincidence technique, it is possible to get the first estimate of the isotopic composition of the particle after about 24 h data acquisition. However, because of the low activity, the determination of ^{240}Pu/^{239}Pu ratio more precisely may require 10 d.

2. PANDA

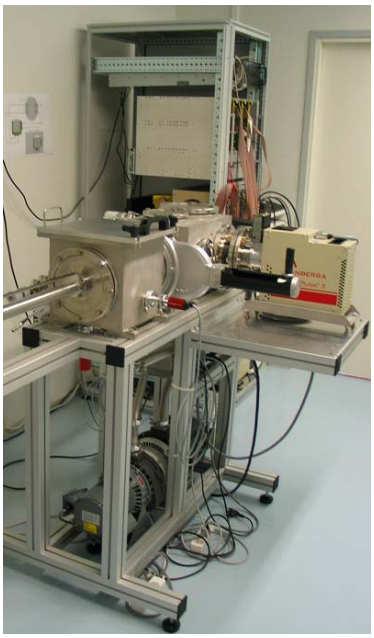
PANDA (Particles And Non-Destructive Analysis) is a development platform rather than a device tailored for specific measurement purposes. It is presented in Fig. 1a. It consists of two vacuum chambers that are connected with a gate valve. The loading chamber is used to pump the sample down into a vacuum. The pump-down from 1 atm to 10^{-9} atm is made slowly using a precision needle valve. When high vacuum is reached the gate valve can be opened and the sample, installed to the tip of a linear feedthrough, is pushed into the measurement chamber. This chamber is always at high

vacuum and it has two measurement positions. The first position, which is already operational, hosts large detectors primarily meant to screen the samples.

Position-sensitive screening for alpha-particle emitting radionuclides is accomplished using a 64×64 mm^2 DSSSD detector that has a pixel size of 2×2 mm^2. Figure 1b demonstrates how the outcome of alpha screening may look like. In the future the screening can be done also for beta emitters. A broad energy germanium detector, with a crystal diameter of 70 mm and a thickness of 21 mm, is installed opposite to the DSSSD detector. The orthogonal distance between the detectors can be varied from a few millimetres to several centimetres. A sample to be screened, such as a swipe, is moved in between the detectors using the linear feedthrough. The centralized event-mode data acquisition system allows versatile views to the collected data. As an example, sophisticated coincidence gating and half-life studies are possible. For more detailed information about PANDA, see [2, 3].

The second measurement position is primarily intended for the studies of individual particles and radiochemically processed samples. It will host small detectors that have the best possible energy resolution. These detectors will be installed to the tip of a second linear feedthrough moving vertically. This way the detectors of measurement position two can be moved to the vicinity of any interesting spot of the sample. Feasibility studies related to these detectors have been finished. Detectors with very thin entrance windows were studied. They provide the capability to detect low energy conversion electrons. Data generated by the detectors of PANDA are stored in a database together with supporting metadata.

(a) (b)



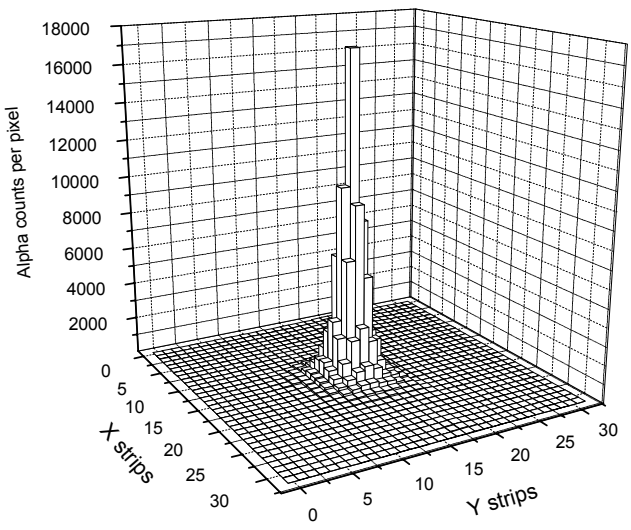


Figure 1. (a) PANDA setup, (b) experimental DSSSD X-Y hit map produced by one particle containing alpha-particle emitting radionuclides.

Using PANDA's particle-specific position information, further analytical studies are made easier. Based on computations and past experience, PANDA is able to detect and locate a 10^{-7} g particle, made of highly enriched uranium (95% ^{235}U and 5% ^{238}U), in about two hours. The detection is based on alpha emission. Notice that more than two hours is needed to characterize the material. In detecting natural uranium PANDA is about an order of magnitude less efficient. For comparison, X-ray fluorescence technique used by the IAEA to screen the swipe samples can detect, locate and elementally identify microgram amounts of U and Pu on the surface of the sample. Analysis of a single swipe takes 4-5 hours [4].

3. PANDA and swipe sample

Figure 1b shows the response of a single particle measured with PANDA. When measuring a swipe the outcome can be completely different. Figure 2 deals with a case where a well known reference material was deposited on a cotton swipe. The produced sample contains 1 µg of uranium and 10 ng of plutonium. The certified atom ratio of $^{240}Pu/^{239}Pu$ is 0.132.

Figure 2a shows the cotton swipe installed to a sample holder. The holder is mounted to the tip of the linear feedthrough which is used to transport the sample between the DSSSD and the HPGe detector. There is a 0.5 µm thick mylar foil in front of the swipe. This is to protect the DSSSD from contamination.

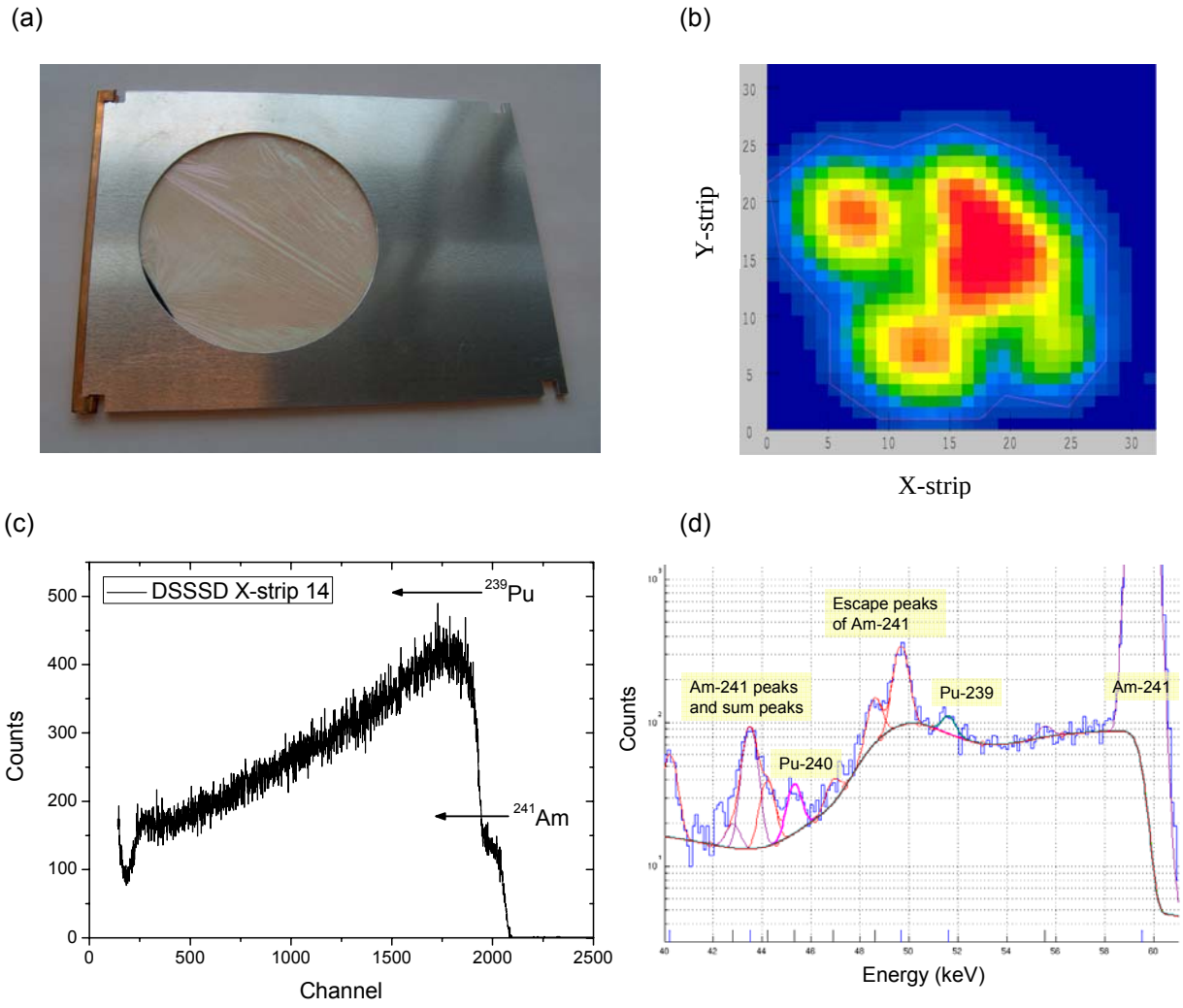


Figure 2. (a) A cotton swipe installed to a sample holder, (b) DSSSD x-y hit map, (c) sample alpha spectrum, (d) AATAMI fit of Pu peaks.

Figure 2b shows an alpha hit map measured with the DSSSD. It is clearly seen that the radioactive material is spread around a large area. Figure 2c shows an alpha spectrum for a single X-strip from the DSSSD. In this alpha spectrum the tailing of peaks towards the lower energies is very clear. This is due to the swipe material and the production method of the sample. Namely, since part of the deposited liquid goes deep into the swipe material and alpha particles lose energy while travelling to the surface of the swipe. Notice that some alpha particles never reach the detector due to the absorption. The cotton swipe is not an ideal material for alpha measurements of this kind. The quality of the alpha spectra can be improved by optimizing the sample collection [5].

Even though the alpha spectra are not of the best quality due to the sample type, alpha signals can be used to create alpha-gated gamma spectra. This results to nearly background free gamma spectrum of alpha decaying nuclides. Figure 2d shows an alpha-gated gamma spectrum of the swipe sample. It shows a fit made with AATAMI-program [6]. This fit includes ^{239}Pu (51.6 keV) and ^{240}Pu (45.2 keV) peaks. Using these transitions the ^{240}Pu/^{239}Pu atom ratio was determined to be 0.12 ± 27 %. In this case ^{241}Am dominates the gamma spectrum and this does not allow a more accurate analysis.

4. Improved sampling methods

The energy loss of the non-penetrating alpha particles is a crucial problem leading to the poor quality of the spectrum. This energy loss results from three absorption mechanisms: absorption to the sample matrix, absorption to the particle itself, and absorption to the material collected on the sample. Even though an ideal sample can only be manufactured with time-consuming methods that increase the risk of cross contamination, a lot can be easily done in field conditions.

Compared to a cotton swipe, far better sample quality can be achieved with more sophisticated sampling methods. The particles of interest penetrate the cotton cloth, and therefore the fibres of the cloth attenuate the radiation significantly. This can be avoided by using a membrane filter as the swipe material. Tests have revealed that the particles of interest cannot penetrate a Fluoropore membrane swipe but the swipe still collects more than 50 % of the particles of interest [5]. A membrane swipe will not solve the problem of self-absorption or absorption to other particles collected. Due to the self-absorption, collection of particles with a diameter larger than 10 µm is not desirable. By using size-separation methods like impactors, only particles with a diameter less than a specific size can be collected. Since only a small mass percentage of the particles in conventional room dust are smaller than 10 µm in diameter, the method also reduces significantly the total number of non-interesting particles [7].

The samples collected with an impactor can have a good quality, but achieving high collection efficiency is difficult. Even an air stream with a speed of 100 m/s is not enough to detach the particles of interest from a hard surface. This problem can be overcome by first swiping the surface with a cotton cloth and then vacuuming the cloth with an impactor [5].

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The Aug 2008 Incident at the IAEA Safeguards Analytical Laboratory Involving a Small Quantity of Pu-240

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Abstract:

A broad range of radioanalytical measurements are carried out in support of nonproliferation treaty verification, nuclear waste processing, environmental monitoring, and national and global security applications at the Safeguards Analytical Laboratory (SAL). A broad range of Certified Reference Materials are utilized to ensure the quality of analyses performed. Some of these reference materials are extremely rare and seldom produced, which may result in considerable storage time before use.

On August 3rd, 2008, an incident later described as INES level 1occurred in a storage safe of SAL. An investigation of this incident strongly suggests that it was triggered by the fracture of a glass vial containing a nitrate solution of plutonium 240. The most likely cause of the breakage is excessive internal pressure arising from the generation of radiolytic gases (primarily hydrogen and oxygen). The rupture of a single vial would not be expected to be a sufficiently energetic event to break adjacent vials, as occurred, or to account for the damage to the tray within the safe on which the vials were stored. It appears likely that the breakage of subsequent vials (a further four) and damage to the tray occurred as a result of a deflagration involving the radiolytic gases released from the first, and perhaps subsequent, vials.

This paper describes the root cause leading to the incident as well as its consequences and the clean-up effort.

Keywords: reference material, plutonium, incident

1. Introduction

The Safeguards Analytical Laboratory performs destructive chemical analysis of nuclear material samples as well as screening and analysis of environmental samples, both taken by inspectors in nuclear facilities as part of the IAEA Safeguards mission. Nuclear material analysis is governed by the accuracies and uncertainties established by the International Target Values (ITV-2000) [1]. These constitute quite ambitious goals and necessitate elaborate quality assurance and quality control measures. Verification of appropriate compliance is done by analysing well characterised specimen, so called Certified Reference Materials (CRMs), where actual results can be compared with certified values. SAL has a large inventory of such nuclear CRMs of various compositions and concentrations to allow these control measurements on a routine basis.

Typically, CRMs in liquid solution are shipped in flame-sealed vials to ensure integrity of the material. Since the radioactive substance causes radiolytic decay of the aqueous solution resulting in buildup of gases, pressure in the sealed vial gradually increases. The SAL incident involved Pu 240 in nitric acid solution, where pressure buildup ultimately caused bursting of the vial and subsequent contamination

of the storage area. Though laboratory safety and filtering systems worked according to design and contained the contamination – nothing was released to the environment – clean-up of the storage area became a major effort, which was successfully completed approximately 5 months after the incident. Due to careful planning, doses to staff could be minimised to negligible levels of external irradiation.

2. Sequence of events

The sequence of events is broken up into various phases. The incident and immediate response measures constitute the initiating event, followed by phase one of the clean-up effort, which addressed other sealed vials potentially at risk from over-pressure. After thorough planning and acquisition of personnel protection equipment, phase two of the clean-up effort focused on decontamination of the storage area and re-fitting the safe for subsequent use.

2.1. Incident

The incident occurred at 02:31 Sunday, Aug 3rd, 2008. Four ampoules containing Pu-240 burst open inside a fire-proof safe in a storeroom at SAL. The air contamination monitor within the storage room automatically notified the IAEA on-call emergency team and the Austrian Research Centre emergency personnel.

The IAEA team remotely checked that no alarm had been issued from the stack exhaust monitor and arrived on site at 03:40. The team first verified that no release to the environment had occurred. All air inside the SAL controlled area is passed through multiple filter banks before release to the atmosphere. The air contamination monitor on the exhaust ventilation air stack showed no measurable levels of Pu-240. Further investigations inside the storage room revealed that the source of air contamination was a securely locked and closed safe. A low level air contamination alarm was also noticed in two other rooms located on the first floor of SAL.

By 8:35, all the filters from the continuous air monitors had been exchanged, analysed and archived. The two laboratories on the first floor where a low level air contamination had been detected were surveyed and found free of any surface contamination above the allowable limit for alpha emitters of 0.4 Bq/cm^2. These laboratories were returned to standard operations on the next day.

The storage room where the incident had occurred was also surveyed, and photographs were made to help with further planning. The room was sealed and secured. These operations finished at 11:00. Thereafter, other decision makers were informed by the acting radiation protection officer about the details of the incident. Later in the afternoon, when sufficient facts were available to confirm the nature and extent of the incident, a summary report was sent to the Austrian authorities and two press releases were formulated and organised by IAEA Headquarters.

2.2. Response action

On the 3rd of August, the air filters of the air contamination monitors were analysed using gamma spectrometry, and the presence of Pu-240 was detected. Pu-240 is a high specific activity alpha emitter which was contained within small glass vials. These were used as standard reference materials. They contain known amounts of nuclear material and are certified by the manufacturer. Within SAL they are used in quality assurance of the laboratory work. Photographs taken from the storage safe soon after the release indicated that one vial containing approximately 0.16 gram of Pu-240 in nitric acid solution had burst; and that four other identical vials had been affected. The five vials containing a total of around 0.8 g of Pu-240 were manufactured in 1993 by KRI, a Russian company. The formation of overpressure inside the vial due to the formation of gases caused by radiolysis since manufacture was suspected to have caused the incident.

The certificate of the manufacturer of the vial does not indicate a limited shelf life or issue a corresponding warning that over-pressurisation could occur. The IAEA, after checking with producers of such reference materials, could not confirm that a similar case had happened before.

The IAEA coordinator of the Network of Analytical Laboratories has issued a warning mail to other laboratories handling similar materials to draw their attention to the possibility of such an incident.

Air and surface contamination measurements showed that the safe in the storage room retained most of the contamination (more than 99 %). Some particles escaped the safe as aerosol during the bursting of the vials and were detected by the continuous air monitoring system. These particles were retained in the two banks of High Efficiency Particulate Air (HEPA) filters installed in the ducting of the room exhaust air system. The accredited testing laboratory at ARC (Austrian Research Centres Seibersdorf) analysed the continuous air monitoring filter of the central exhaust stack 1 day after the incident and found no alpha contamination on the filter due to the incident.

The laboratory contamination was assessed by handheld monitors, swipe samples and deposition measurements (detection of the deposition of airborne particles into beakers). The storage room was found to have slightly elevated alpha surface contamination directly in front of the safe measured both directly and also by high sensitive Inductively Coupled Plasma Mass Spectrometry (ICP-MS). The remainder of the laboratory showed surface contamination values for alpha emitters of less than 0.4 Bq/cm^2. Except for the storage room, normal operations were resumed in SAL on the next day.

The ARC (Austrian Research Centres) was requested to carry out environmental measurements relevant to this incident. Swipe samples were taken from the roof of SAL (location of the exhaust stack), and also rain water draining from the SAL and other roofs (there was heavy rain on the afternoon of Monday, Aug 4^{th} at Seibersdorf) was sampled and tested for Pu-240. In addition, samples of soil, vegetation and water from reference measurement points in the environment were collected. Air samples were made using the continuous (ARC) aerosol sampler and surface water were collected from the ARC fire water pond. All samples evaluated indicate values within their normal background range, determined over many years of environmental sampling campaigns. The final ARC report released to the IAEA on 2008/09/08 states 'The measured values of activity of the "incident" in the environmental samples do not deviate from the routinely performed environmental monitoring results.' Based on the results from the ARC accredited laboratory the initial assumption that the incident released no radioactive material into the environment was confirmed.

In conclusion, the safety systems and the emergency team activities resulted in the safe containment of the contamination and prevented any release of radioactive substances to the environment. The impact on the environment has therefore been confirmed as zero. The impact on current laboratory operations has resulted in minor delays of analytical work. The clean-up and decontamination work has required many man-hours of work, and has resulted in substantial cost to the IAEA.

Personnel monitoring of doses to first responders and clean-up staff resulting from this incident due to Pu-240 have been found to be below the detection limit as confirmed by alpha spectrometry techniques applied to urine analysis. A comprehensive dose assessment has been provided upon completion of phase 2 of the clean-up effort.

The incident has been rated as an Anomaly (level 1 incident) on the International Nuclear Event Scale. (INES). The INES scale is a seven level scale, used internationally by nuclear facilities, to categorize the seriousness of accidents and incidents.

2.3. Clean-up

The full inventory of reference materials and samples was examined to identify any similar materials which could be subject to a similar pressure build-up. Five additional vials containing 0.2 gram of U-235/Pu-240 mixture were identified and needed to be secured as a first priority into specially modified Type B radioactive material transport containers prior to any further action being taken. These containers are equipped with provision for pressure release through a HEPA filter should the vial burst within the container. Personnel protective equipment for contaminated environments as well as protection against bursting vials was employed during the phase 1 clean-up which comprised opening the safe and transferring the identified samples to the Type B containers

Based on a comprehensive, approved work plan, the phase 1 clean-up was successfully completed on 2008/08/22. Swipe samples were taken from the surfaces inside the safe to allow the planning of the next phase. ICP-MS measurements of body and nasal swipes as well as standard urine analysis showed that no detectable internal doses had occurred.

After the phase 1 clean-up, the floor of the storage room was decontaminated to allow the preparation for the phase 2 work, which involved the construction of temporary enclosures (tents) around the safe
A detailed work plan for phase 2 clean-up (see fig. 1) was prepared in cooperation with Austrian experts and approved internally by the radiation safety regulator. The procurement of personnel protective and supportive equipment was initiated in September 2008, but long delivery times delayed further work until beginning of December.

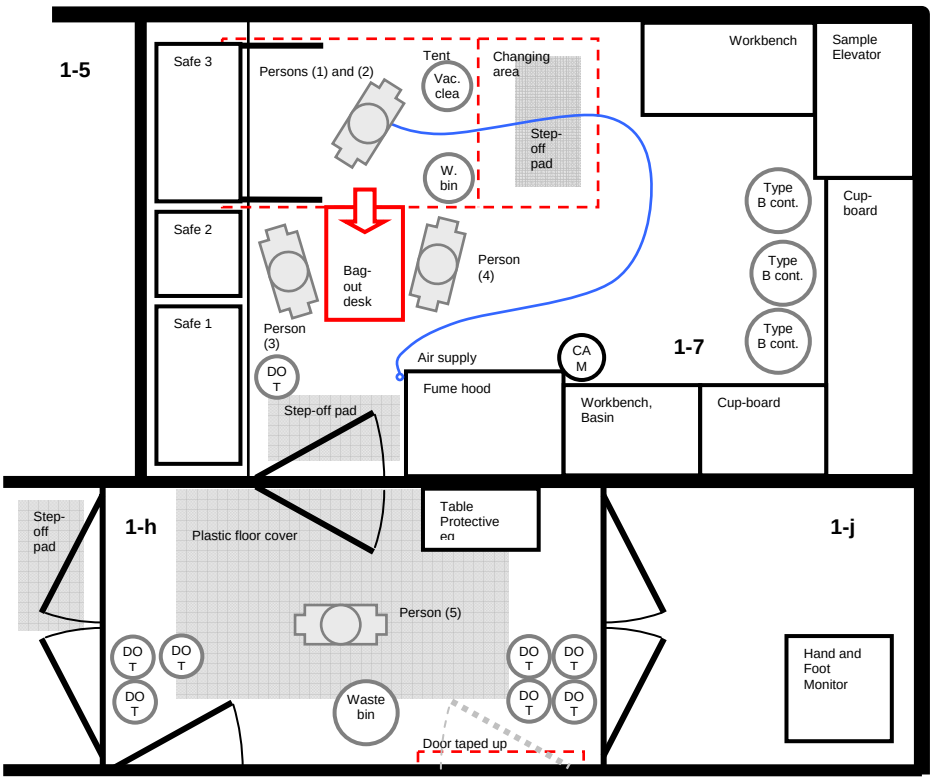

Fig. 1: Schematic arrangement for phase 2 clean-up in the storage room

The phase 2 clean-up commenced on 2008/12/02 with the removal of all radioactive samples and materials from safe 3. One person in a ventilated, positive pressured suit worked inside a protective tent in front of the safe and passed material through an access port for subsequent double-bagging and sealing (see fig. 2). The drawer of the safe where most of the Pu-240 was deposited was then placed into a glove box where the recovery of the remaining Pu-240 will take place.

Fig. 2: Bag-out of material contaminated in the incident from within the protective enclosure
As it was decided to retain safe 3 for future use, this phase of the clean-up also involved the decontamination of safe 3 itself and the disposal to active waste of the contaminated removable parts (trays and sliding mechanism). Permanent markings have been applied to safe 3 to indicate that the

interior, non-accessible components of the safe are contaminated. All inner surfaces of safe 3 have been sealed using special decontaminable paint, and a new insert provides mechanical support to shelves and sliding trays, installed by the on-site mechanical workshop. No removable contamination remains on the internal and external surfaces of the safe. Finally, the room itself was thoroughly decontaminated. A survey was conducted to verify the absence of surface contamination in the storage room and safe 3, after which the room was restored to its normal functions. Phase 2 was successfully concluded on 2009/01/09.

Further activities (phase 3) will focus on the material secured in phase 1, at present stored in modified Type B containers. It will involve a safe procedure of releasing the excess gas pressure inside the vials and re-packaging the material.

All activities have been and are being planned with the support of internal and external experts under the approval of the radiation safety regulator. They also involve the use of appropriate personnel protective equipment and extensive personnel monitoring for incurred radiation doses.

3. Root cause

3.1. Background

The decomposition of material through energy from radioactive decay is well known. After the incident at SAL on 3 August 2008, an extensive search of literature was made to track various references and check for similar occurrences in the past. Though multiple reports were found on this issue in general and some reports on the failure of packages, none documented a case of certified reference materials causing problems in storage. This paper therefore presents an overview of literature findings and derives the formalism by which SAL calculates henceforth the generation of hydrolytic gases produced during storage in order to guide future safety assessments.

Radiolysis is also mentioned in the IAEA publication “Safe handling and storage of Plutonium” [2] and the DOE standard “Guide of good practices for occupational radiation protection in Plutonium facilities” [3]. Neither these publications nor the Safety Analysis Report of SAL mention the risk to the integrity of sealed vials in storage, or that reference materials such as those stored in SAL could burst due to the resulting pressure build-up caused by radiolysis over a number of years.

Considering the damage to the safe, it also appears that the released H_2 may have recombined energetically with oxygen, which seems consistent with estimates of the total energy released from the produced amounts of H_2 and O_2. This strong reaction affected the remaining four ampoules causing their bursting and possible deflagration in turn, and minor physical damage to the safe. However, it remains unclear which process caused the ignition and deflagration of the gas mixture.

Another task – not discussed in this paper – remains with respect to the material in pressurised vials taken in safe storage. Special procedures will have to be applied to recondition these materials.

3.2. Relevant Literature

Radiolytic decomposition of actinide-containing nitric acid generates a complex mixture of gases. among them NO_2, N_2O, N_2, NO, O_2, and H_2. Precursor species and the mechanisms by which the precursors react to form gases were reviewed many years ago [4]. In a particularly useful paper that appeared in 1973, Weiss and Pietri [5] presented an easy-to-use equation that predicts the rate of hydrogen generation in nitric acid solutions of plutonium. The Weiss-Pietri equation predicts the rate of hydrogen generation in moles per liter of hydrogen per day. Kazanjian and Horrel [6] further expanded this model to use exponential type equations for the prediction of hydrogen generation. More recent evaluations by Kuno et.al. [7] generally confirmed the original findings with some variability, where experimentally determined values are plotted as a function of acid concentration (see also Burns [8] and Sheppard [9]). A comprehensive summary is provided in Silver and Anderson [10], based on an earlier review by Silver [11]. Recent studies also investigate the influence of other constituents of the solution on the radiolysis of plutonium (e.g. Rance and Zilberman [12]).

A Los Alamos report deals with various PuO and Pu metal storage package failures [13], mainly sealed solid waste containers. Various occurrences of container failure are listed, partly due to

pressurization. Table 2 of [13] lists 7 case studies involving mostly plutonium scrap, attributing part of the processes to oxidation and/or corrosion. Many papers draw attention to the fact that a combustible mixture of gases may be generated, leading to explosive recombination of hydrogen and oxygen.

No cases were found in the literature regarding incidents involving reference materials.

3.3. Formalism of Calculations

3.3.1. Background

Based on the literature review in the previous section, the predominant process is radiolytic decomposition of the aqueous solution, yielding primarily hydrogen and oxygen with the ratio 2 : 1. Many influence factors govern this process, specifically

- type of acid solution (this derivation concentrates on nitric acid solutions)
- concentration of acidic solution (molarity)
- concentration of Pu in g/L
- isotopic mixture of Pu and chemical form (Pu(IV) or Pu(VI))
- time elapsed since production/sealing of vial

For actual risk assessment of pressurization of storage vessels the geometry of the glass vial, the degree of filling and the headspace, i.e. empty space at the top of the vial, need to be known.

Detailed treatments in literature consider interaction of gaseous and liquid components and recombination of hydrolytic gases after generation. This simple-minded treatment does not attempt to account for these effects. Experimental studies derive a "hydrogen yield parameter" $G(H_2)$, characterising the average number of hydrogen molecules produced by 100 eV of energy absorbed for the specific environment. Though literature based estimates exhibit a degree of variation in this parameter, there seems to be sufficient convergence to base the hydrogen (and corresponding oxygen) generation estimates on these published figures.

3.3.2. Calculations

The present estimates are based on the original derivation of Weiss and Pietri due to its self-explanatory approach. An essential parameter influencing hydrogen generation is nitric acid concentration. Experimental results for $G(H_2)$ are reflected in fig. 3.

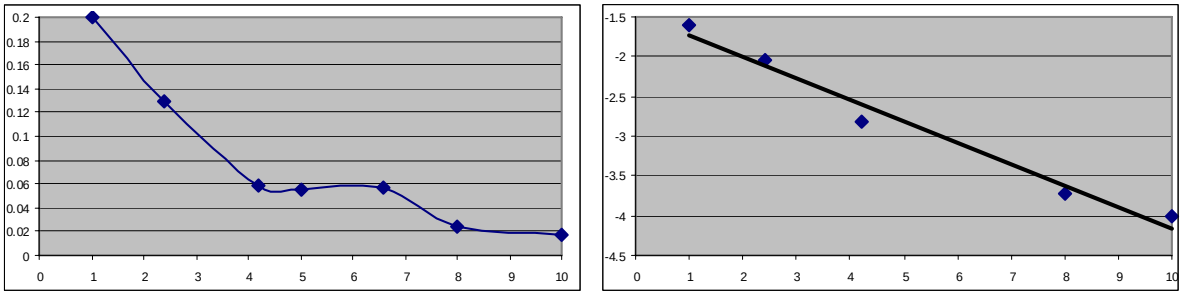

Fig. 3: $G(H_2)$ value as a function of molarity of nitric acid concentration. The diagram to the left shows original values from the table in Weiss and Pietri, the diagram to the right is the logarithmic and linearised version used to derive a formula for $G(H_2)$ based on molarity of nitric acid

Over the concentration range of 1-molar to 10-molar, the corresponding $G(H_2)$ value varies over one order magnitude. These values are consistent with other studies such as [7] or [8].

Another key parameter is the specific radioisotope involved. Since alpha-decay is the predominant process and the energy of alpha decay is relatively stable at 5.5 MeV, the concentration and specific activity of the radioisotope(s) are the remaining components factoring into the equation (disregarding all other minor effects). Table 1 lists the relevant properties of the most important plutonium isotopes considered for certified reference materials. For radioisotopes of (comparably) shorter half life (such as Pu-241 and Cm-242) the decay daughters are also specified (Am-241 for ß decay of Pu-241, Pu-238 for α-decay of Cm-242). The daughter products of Pu-238 do not add significant decay heat.

Radioisotope	decay	Half-life [years]	λ [1/a]	Spec.act. Bq/g	Decay W/g total	Decay W/g alpha

Pu-238	α	87.76	7.90E-03	6.34E+11	5.67E-01	5.66E-01
Pu-239	α	2.41E+04	2.88E-05	2.30E+09	1.93E-03	1.92E-03
Pu-240	α	6587.5	1.05E-04	8.40E+09	7.06E-03	7.04E-03
Pu-241	ß	14.41	4.81E-02	3.81E+12	3.27E-03	7.32E-05
Pu-242	α	3.74E+05	1.85E-06	1.46E+08	1.17E-04	1.16E-04
Am-241	α	433	1.60E-03	1.27E+11		1.13E-01
U-233	α	1.59E+05	4.36E-06	3.56E+08	2.80E-04	2.80E-04
Cm-242	α	4.46E-01	1.55E+00	1.23E+14	1.22E+02	1.22E+02
Cm-244	α	1.81E+01	3.83E-02	2.99E+12	2.83E+00	2.82E+00

Table 1: Properties of radioisotopes used in the calculations (from Karlsruher Nuklidkarte [14])

For observation periods short in comparison to the half life of the radioisotope in question, decay effects may safely be disregarded. However, for Cm-242 and Pu-241 (and Pu-238) this assumption does not hold. Therefore, not only the varying amount of the mother nuclide, but also the build-up of the daughter product must be considered for assessments exceeding a few years. This is based on the well-known equation for radioactive decay:

$$N(t) = N(0) \cdot e^{-\lambda \cdot t}$$

Substituting mass in gram for N (number of atoms) and integrating over the desired duration in years yields an estimate of mass-years of material

$$\int_0^T m(0) \cdot e^{-\lambda \cdot t} dt \Rightarrow m(0) \cdot (-1/\lambda) \cdot [e^{-\lambda \cdot T} - 1]$$

for duration T in years (the decay constant λ must also be expressed in years^{-1}). The term mass-years times alpha energy output ("decay heat") is a measure for total (alpha) energy available over the time period of T years. Hydrogen generation may therefore be expressed as

$$mL(H_2) = \frac{m}{\lambda} \cdot [1 - e^{-\lambda \cdot T}] \cdot P \cdot \frac{365.25 \cdot 86400}{1.6 \cdot 10^{-19}} \cdot \frac{G(H_2)}{100} \cdot \frac{22.41 \cdot 1000}{6.023 \cdot 10^{23}}$$

m	initial mass of radioisotope in g
λ	decay constant of specific radioisotope in years^{-1}
P	alpha decay heat in W/g
$G(H_2)$	hydrogen generation constant as derived from nitric acid environment
T	duration in years

The above equation converts W/g of alpha decay heat to total energy per year in eV and further converts the produced gas to moles and the corresponding mole-volume to mL.

$$W/g = \frac{J}{s \cdot g} \Rightarrow \frac{J \cdot 365.25 \cdot 86400}{1.6 \cdot 10^{-19} \cdot s \cdot g} [eV / year \cdot g]$$

For daughter nuclides of radioactive decay the process of deriving alpha induced hydrogen generation is identical, however the derivation of mass-years follows the integral equation for build-up and decay of the daughter nuclide

$$\int_0^T m_D(t) \cdot dt = m_M(0) \cdot \frac{\lambda_M}{\lambda_D - \lambda_M} \cdot (e^{-\lambda_M \cdot T} - e^{-\lambda_D \cdot T})$$

m_M	initial mass [g] of the mother nuclide
m_D	mass [g] of the daughter nuclide after time T
λ_M	decay constant [years^{-1}] of the mother nuclide
λ_D	decay constant [years^{-1}] of the daughter nuclide
T	duration in years

3.4. Examples

A simple Excel-spreadsheet has been developed to calculate hydrogen yield based on total mass of radioisotope, specification of weight percent of isotope mixture, molarity of nitric acid solution and time in years.

Complex calculation with decay correction (and daughters Am-241 / Pu-241 and Pu-238 / Cm-242) according to alpha isotopic power and Weiss/Petri G(H2)

mass [g]	M HNO3	time [yr]	wt% Pu-238	wt% Pu-239	wt% Pu-240	wt% Pu-241	wt% Pu-242	wt% U-233	wt% Cm-242	wt% Cm-244	mL H2
0.16	**1.5**	**15**	**0**	**4**	**95**	**1**	**0**	**0**	**0**	**0**	**186.029**

The above figure is a reproduction of this spreadsheet, highlighting a calculation for a total mass of **0.16 g** of plutonium in **1.5 molar** nitric acid solution, containing a mixture of **95 weight-% of Pu-240**, 4% of Pu-239 and 1% of Pu-241. The resultant hydrogen after **15 years** amounts to approximately 186 mL (oxygen yield ~ 93 mL). For 30 mL headspace this results in 9+1 bar pressure.

It must be emphasised that this is an approximate assessment of hydrogen and oxygen generation under simplified assumptions, disregarding more complex interactions. For the purpose of general safety assessments the uncertainty of this derivation in the order of ± 20% might be sufficient, erring more on the safe side by disregarding recombination effects. However, to calculate pressurisation of vessels the headspace of the vials needs to be known to derive pressure estimates.

4. Lessons learned

An internal independent committee was created to identify the lessons learned. Their executive summary and the full internal report of the high level working group has been submitted to IAEA management.

A number of areas for further consideration or actions have been identified. In broad terms these can be divided into those specific to the actual incident and the emergency response arrangements, and those of wider significance of a more far-reaching nature. The SAL-specific actions include:

- Identification of other plutonium samples which may be under significant internal pressure;
- Reviewing the SAL inventory against foreseeable requirements;
- Considering the stabilisation of high value but infrequently used material;
- Revising various aspects of the SAL Emergency Response Handbook;
- Reviewing the SAL Safety Analysis Report to re-evaluate the most significant, credible risks;
- Ensuring that responsibilities for incident and emergency response are clearly assigned.

Some changes to the Emergency Plan were immediately implemented, such as an expansion of the list of telephone numbers on the “On-call” list as well as a risk assessment based on radiolysis pressure build-up for the other materials in storage. To assure reliable storage of reference materials (which may be kept in storage for considerable periods of time) producers will be requested to provide documentation on safe storage periods or other necessary precautions.

After review of all suggestions and recommendations, as part of the regulatory process of renewal of the authorisation for SAL operations, corresponding changes have been implemented in standard operating procedures, the emergency plan(s), and the laboratory quality management system.

Although the safety systems worked well to contain the incident, similar incidents must be prevented in the future. A corresponding warning has already been issued to the Network of Analytical Laboratories world-wide. Further actions for SAL comprise:

- Suspension of all work at SAL involving samples containing alpha emitters until the overpressure safety of the ampoule has been assured.
- Setting up a calculation algorithm to calculate the overpressure in each sample vial at SAL. The algorithm relates the activity, time of storage, gas space inside the vial, pH, matrix and other factors to calculate the quantity of hydrogen formed and the resulting overpressure.

- Update of the SAL Safety Analysis Report as part of the regulatory process of renewal of the authorisation for SAL operations to account for line management response in case of emergencies.
- Storage of potentially hazardous materials in specially modified Type B containers until further treatment.
- Review of all IAEA plutonium storage procedures, including those for metallic plutonium.
- Provision of safety information to external laboratories where similar samples are kept.
- Recommendations to suppliers of such samples to provide adequate safety information.

4. Conclusion

The incident at SAL has demonstrated the importance of assessing the hazard of storage of radioactive certified reference materials in sealed glass ampoules and corresponding safety assessments regarding the long term integrity thereof. The incident also highlighted that good laboratory design foresees multiple layers of containment, effectively prohibiting a release to the environment, even in the case of minor incidents. Well-established and well-practiced emergency procedures assist in mitigating the consequences of such an incident, whereas well-trained Health Physics staff are an indispensable asset in conducting decontamination and clean-up. This was ultimately confirmed by the absence of internal doses of first responders and clean-up crew attributable to the incident.

There are many important “lessons learned”, which will be captured, along with the circumstances of the incident and the subsequent activities, in a detailed technical report to be published by the IAEA. However, three important conclusions of wider significance may already be drawn from this event:

(1) It is highly recommended that manufacturers of certified reference materials provide indications of the safe storage period of their products, if hermetically sealed containers (flame-sealed vials) are used. Physical dimensions and filling amounts should leave sufficient free volume to ensure long shelf life. Corresponding warnings, drawing attention to the risk of pressure build-up and subsequent containment failure, should be included in the supporting documentation.

(2) With the phenomenon of radiolysis well known, this may easily be applied to material storage issues as described above. A simple derivation has been accomplished to assess total hydrogen produced over prolonged periods, based on literature values of hydrogen generated for Pu and other critical radioisotopes in specific nitric acid environments and total alpha energy (decay corrected). Such algorithms should be applied to existing inventories on a routine basis.

(3) Laboratory design has a significant impact on the capability to contain minor incidents. However, many nuclear laboratories are quite aged facilities with corresponding maintenance problems affecting the technical infrastructure. Ensuring that all containment infrastructure (e.g. ventilation and filtering systems) are maintained at full design capability is a non-negotiable safety precautions, as evidenced in the case of the SAL incident. Addressing this concern may necessitate major investments to ensure safe operations of a nuclear laboratory.

5. Acknowledgements

The IAEA acknowledges the support received from internal and external experts from Austria and from the Austrian Research Centers and their radiation protection team. The IAEA also acknowledges the constructive dialogue with the Federal Ministry of Agriculture, Forestry, Environment and Water Management of the Republic of Austria.

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SESSION 7

CONTAINMENT AND SURVEILLANCE – II

The development and implementation of a safeguards approach for the export of fabricated MOX fuel from France to Japan

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Abstract:

The use of mixed Uranium and Plutonium oxide (MOX) fuel assemblies is a way to use the Plutonium extracted during reprocessing of irradiated fuel for power production. In this respect the MELOX plant is the main producer of MOX fuel assemblies worldwide. The plant is safeguarded by EURATOM since the beginning of operation and the safeguards scheme in place is based on a combination of Containment and Surveillance (C/S) systems coupled with unattended measurement systems, branching of selected operator's flow measurement equipment and interim physical verifications. Early in 2008 the plant started the production of MOX fuel assemblies for use in Japanese power reactors and the first transport of the fuel assemblies off the site took place in August 2008. The fuel assemblies are sent from MELOX by road transport in FS65 containers to La Hague where they are reconditioned into TN12 sea containers for sea transport to Japan. Since safeguards verifications at reactors are intrusive and resource demanding, the IAEA wished to carry out the necessary verifications at MELOX and to make use of the results of the verifications performed by EURATOM in the process during the fuel assembly fabrication, while maintaining continuity of knowledge during the shipment. The close cooperation between the plant operator, IAEA and EURATOM has demonstrated the benefit both in terms of efficiency of resources used and limited intrusiveness of safeguards activities.

Keywords: IAEA, Co-operation, MOX, Verifications

1. Introduction

MOX fuel is considered amongst the most sensitive material in nuclear safeguards. To establish a detailed knowledge on quantities and composition of nuclear materials contained in fuel assemblies, the fuel fabrication plants are the best place to carry out safeguards verifications. Safeguards measurements at reactor sites are more intrusive to plant operations, include the risk of damages and are normally of lower accuracy. Especially for fresh MOX material it is essential to have a high degree of confidence in the verifications and acceptable measurement conditions to allow for the accuracy required. For MOX fuel produced at the MELOX plant in France for Japanese customers, it was decided to support the IAEA with their verification requirements by sharing safeguards data, support additional requirements and allowing Quality Assurance (QA) checks to be carried out.

EURATOM Safeguards is applied to all civil nuclear installations in France and the EURATOM safeguards scheme at MELOX is one of the most advanced, consisting of several safeguards instruments running in unattended mode and evaluations done with a high degree of automatisation. These verifications allow EURATOM Safeguards to draw comprehensive safeguards conclusions with a high level of confidence and to the accuracy level required.

As in other Nuclear Weapon States (NWS), the IAEA applies safeguards in France on a voluntary offer basis. In MELOX a separate Material Balance Area (MBA) has been created for exports to Non Nuclear Weapon States (NNWS), which is designated by the IAEA.

To avoid costly, cumbersome and intrusive verifications on arrival of the fuel assemblies in Japan, the IAEA decided to make best use of safeguards verifications at MELOX and to maintain continuity of knowledge during the transfer and repacking operations from land to sea type containers, that take place in La Hague.

The safeguards arrangements require some flexibility of all parties involved. The safeguards organisations do not have the resources to be present at the installations for prolonged periods of time to observe handling operations. Extended use of surveillance systems had to be made and the operator's handling of tamper-proof sealing equipment was essential to achieve these safeguards goals.

Especially for the handling operations at the La Hague site, when the fuel assemblies are received and repackaged from land transport to sea transport containers, a secure and redundant CIS system has been installed that allows the operator to handle electronic seals under surveillance. Safeguards inspectors later check these manipulations and confirm the continuity of knowledge without being on the critical path of the operations.

These arrangements are an example of how efficient and effective safeguards can be performed without excessive resource requirements, if all parties involved show some flexibility and co-operation.

2. MELOX

2.1. EURATOM Safeguards scheme at MELOX

The MELOX plant, located at the Marcoule site in the south of France, is in operation since 1994 and has been from early project stages one of the focal installations for safeguards in Europe due to the sensitivity of the material processed. The plant has increased its throughput from initially 100 tonnes to 145 tonnes in 2003, and recently further up to 195 tonnes.

EURATOM has adopted a tailor made approach taking into account the specific nature of the plant. The sensitivity of the direct use Plutonium material and the high throughput of the plant require a robust safeguards system; that nevertheless allows a continuous operation of the production process. Because of limited inspection resources, these systems are directly integrated into the process flows, run in unattended mode and allow, together with the related data collection and evaluation tools, for an on line process monitoring. Together with appropriate Containment and Surveillance (C&S) measures the intrusiveness of safeguards activities to plant operations has been minimised and allows for a continuous operation of the plant between Physical Inventory Takings without the need to compromise production targets because of safeguards verifications.

The safeguards instrumentation allows the inspectors to monitor the main material flows in the plant and to compare them with the operating data submitted on a daily basis. Since significant data volumes need to be processed the a computer application called MIDAS is used to load these operating data, check them for internal consistency and compare them with the signals of the safeguards instrumentation running in unattended mode. The plant processes are therefore mainly monitored from two sides, the very detailed plant control at the signal level and the overall consistency checks of the operating records and accountancy records with monthly declarations. This results in a very detailed insight into plant operations and the performance of the operator's Nuclear Material Accountancy and Control (NMAC) system.

MOX fuel assemblies are produced and stored at the MELOX facility within the Material Balance Area (MBA) FML3 prior their shipment.

The fuel assembly store is under EURATOM containment and surveillance system and detailed knowledge of the nuclear material contained is established through a branching of operator's

equipment and independent, unattended neutron and gamma measurements performed during the fabrication process.

A separate MBA (FML2) has been created to deal with the shipment of fuel assemblies to Non Nuclear Weapon States (NNWS). The borderline between the two MBAs is at the exit of the fuel assembly store and MBA FML2 includes the whole shipment area, which is used for the packing and shipment of fuel assemblies.

Continuity of knowledge on the fuel assemblies is maintained throughout the packing and shipment operations by a combination of video surveillance systems and seals.

In order to enable the IAEA to draw their own conclusions, an agreed set of data derived from the EURATOM NDA equipment and surveillance systems installed within MBA FML3 and relevant to fuel assemblies shipped to Japan, is made available to the IAEA. In addition it was agreed that the IAEA could perform NDA measurements on randomly selected fuel assemblies upon request for QA purposes. The maximum number of these measurements would normally not exceed 10% of the total number of fuel assemblies shipped during the year.

All movements of fuel assemblies from the fuel assembly store to the transport containers are under full surveillance by EURATOM cameras connected to a centralised recording system (FAST) installed in the on site EURATOM office. The video signal of the relevant video cameras has also been connected to a redundant IAEA recording system (QDIS).

The fuel assemblies are transferred vertically by crane from the fuel assembly store to the “fuel holder” loading bay and identified by their serial number with a specific surveillance camera. Each fuel assembly is placed into an individual fuel holder.

For PWR fuel assemblies, the fuel holder is placed around and locked to the head of the fuel assembly by a mechanism in the fuel holder end cap. For BWR fuel assemblies, the fuel holder is placed around the fuel assembly and closed by means of bolts in the end cap.

The fuel assembly in its fuel holder is then transferred under camera surveillance to the transport container (FS 65 type). If required by the IAEA for QA purposes, the fuel assembly in its holder is placed into the IAEA neutron collar and gamma detector located in the transport container loading pit and neutron and gamma measurements for partial defects and active length are performed with IAEA inspectors' presence.

After the above IAEA verification or directly after the loading into the fuel holder, if the verification is not required, the fuel assembly in its fuel holder is inserted into the transport container. After loading, the transport container is closed, removed from the loading stand and placed under surveillance horizontally in the intermediate transport container storage area. Once all containers for a shipment are accumulated in the intermediate storage area, each of them is sealed with a COBRA and an electronic (EOSS or VACOSS) seal. After sealing, the containers may be removed from the camera surveillance area and loaded into the road transport container without inspectors' presence. There is no requirement for inspectors’ presence or sealing of the road transport containers provided that the integrity of the seals on the fuel assembly transport containers is maintained.

2.2. Adaptation of existing systems at MELOX

Following detailed discussions between the operator, the state authorities, the IAEA and EURATOM, it was decided to install additional cameras to fully cover the transport path from the EURATOM neutron collar measurement system through the Fuel Assembly Store, the packing area to the shipping area in order to avoid the continuous presence of inspectors during the loading operations.

It was also agreed that relevant data including regular updates of the planned fabrication and loading schedule would be sent to an email account to both EURATOM and the IAEA. These agreements included also a notification by email of the loading date of the shipping containers at the latest 48 hours before the start of the operation.

2.3. Inspection activities at MELOX

Upon arrival on site, seals are placed on fuel assembly containers, which have been already loaded and temporarily stored under video surveillance. A set of documents, including relevant safeguards data, such as fuel assembly identification, Plutonium and Uranium content and isotopic composition, is transmitted to the inspectors. These data are later used for the reconciliation with the video review results and the measurement results acquired through the EURATOM measurement systems during the fuel assembly fabrication process (FML3). Inspectors then perform the joint team review for the period starting from the previous inspection until their seals placing, for the relevant fuel assemblies
.
Normal shipment inspection activities with the IAEA at MELOX include:

- Sealing of loaded transport containers
- Video review from the previous inspection until the sealing of the Fuel Assembly transport containers by inspectors.
- Verification of accountancy records and collection of EURATOM measurement data for the IAEA (only during quarterly inspection).

One shipment represents 4 shipping containers containing either one PWR assembly or two BWR assemblies each. Normally two inspection man days are needed per inspection and per inspectorate to perform these verification activities.

2.4. Anomaly resolution at MELOX

If the surveillance reviews or the evaluation of instrument data show inconsistencies with the operator's declarations or are inconclusive, follow up or remedial actions will be taken depending on the severity of the findings and if the operator caused or could have been aware of any equipment failures.

If the C/S review is inconclusive upstream of the entry to the fuel assembly store a limited number of fuel assemblies will be requested to be re-verified using the neutron collar measurement system.

The IAEA instrumentation at the transport container loading pit will be used to recover from any C/S failure or anomaly detected between the fuel assembly store and the loading of the transport container.

After the loading of the transport container, recovery verifications from C/S failures will be carried out at La Hague or depending on the time of detection at the receiving installation in Japan. Any results from re-verification measurements carried out by the IAEA in Japan will be communicated to EURATOM.

3. LA HAGUE

3.1. Description of operations at La Hague

Once road transport containers shipped from MELOX, arrive in Cherbourg, the MOX fuel assemblies must be transferred to sea transport containers before they can continue their journey. These operations are performed in La Hague.

The FS 65 containers are removed from the road transport containers (CBT 06) and transferred to the pond area. The fuel assemblies with their fuel holders are then removed from the FS 65 containers and transferred to TN12 type sea transport containers.

Once full, the TN12 containers are closed and transferred to the receipt / exit area where shock absorbers are mounted to the flasks to complete the export preparations.

3.2. Driving Concepts for the Safeguards activities at La Hague

3.2.1. Continuity of Knowledge

As there are no means at La Hague to accurately determine the quality and quantity of nuclear material present in fresh MOX assemblies, the safeguards approach must rely on the knowledge acquired at the shipping facility.

Consequently, the safeguards approach at La Hague is based on maintaining the continuity of this knowledge throughout the handling operations.

This is achieved through:

- The sealing of the road transport containers at MELOX and for the duration of the transport between MELOX and La Hague (EOSS or VACOSS electronic seal and COBRA seal),
- The maintenance of the nuclear material under surveillance during all reconditioning operations, and
- The application of seals on the sea transport containers (Common metal and COBRA seals).

3.2.2. Optimisation of Resources

In the years 1999-2000, a 1st series of fresh MOX assemblies coming from FBFC Belgium had already been reconditioned in La Hague and exported to Japan. At that time the implemented safeguards approach, while already relying on surveillance measures, was based on a rather intensive on-site presence of joint-team inspectors during these reconditioning operations.

However for the new series of exports, a focus was to optimise resources by minimising inspectors' presence during handling operations and also to minimise safeguards intrusiveness to plant operations.

To this end, solutions were sought for to disconnect as much as possible inspectors' presence from operator's reconditioning operations.

The following options were explored:

- Delegation under condition of certain handling tasks to the operator;
- Collection of recorded data from safeguards equipment and evaluation of these data "a posteriori", i.e. after the sealing of TN 12 containers by the Joint-Team.

3.3. Safeguards implementation at La Hague

3.3.1 Methodology

The Safeguards implementation scheme was designed in full co-operation not only with the IAEA, but also with the operator and the state authorities. All safeguards aspects with a possible impact on operations or the operator were treated in a fully transparent approach (proposed, discussed and agreed). Envisaged solutions were presented to the operator and accepted following the demonstration by the inspectorates that proposals were sound, relied on secure safeguard equipment and that all situations departing from normal operations (e.g. change in operators' handling procedures, failure of safeguards equipment) had been considered, their consequences evaluated and appropriate responses agreed.

3.3.2. Delegation of Tasks to the Operator

In order to disconnect handling operations from the presence of inspectors on site, e.g. when the FS 65 road transport containers have to be unloaded, the task to detach the seals, placed at MELOX on the FS 65 containers, has been delegated to the AREVA operator. These operations are performed in the receipt area only when the CBT 06/FS 65 transport containers are under safeguards camera surveillance.

This concerns the following actions:

- COBRA picture taking;
- Downloading of recorded VACOSS information to an ALIS camera;
- Detaching of COBRA seals and opening of VACOSS seals.

In case of an external integrity anomaly is detected by the operator on COBRA seals applied on a received FS 65 transport container, both inspectorates are immediately notified by the operator and arrangements are made so that inspectors can be present on site within the next 36 hours. In the meantime and in order to minimise consequences on the operator, the concerned FS 65 container can still be transferred to the reconditioning area, but must not be opened and the remaining seals must not be detached before inspectors are present.

3.3.3. Inspection Scheme at La Hague

The inspection scheme applied at La Hague is described in figure 1 below.

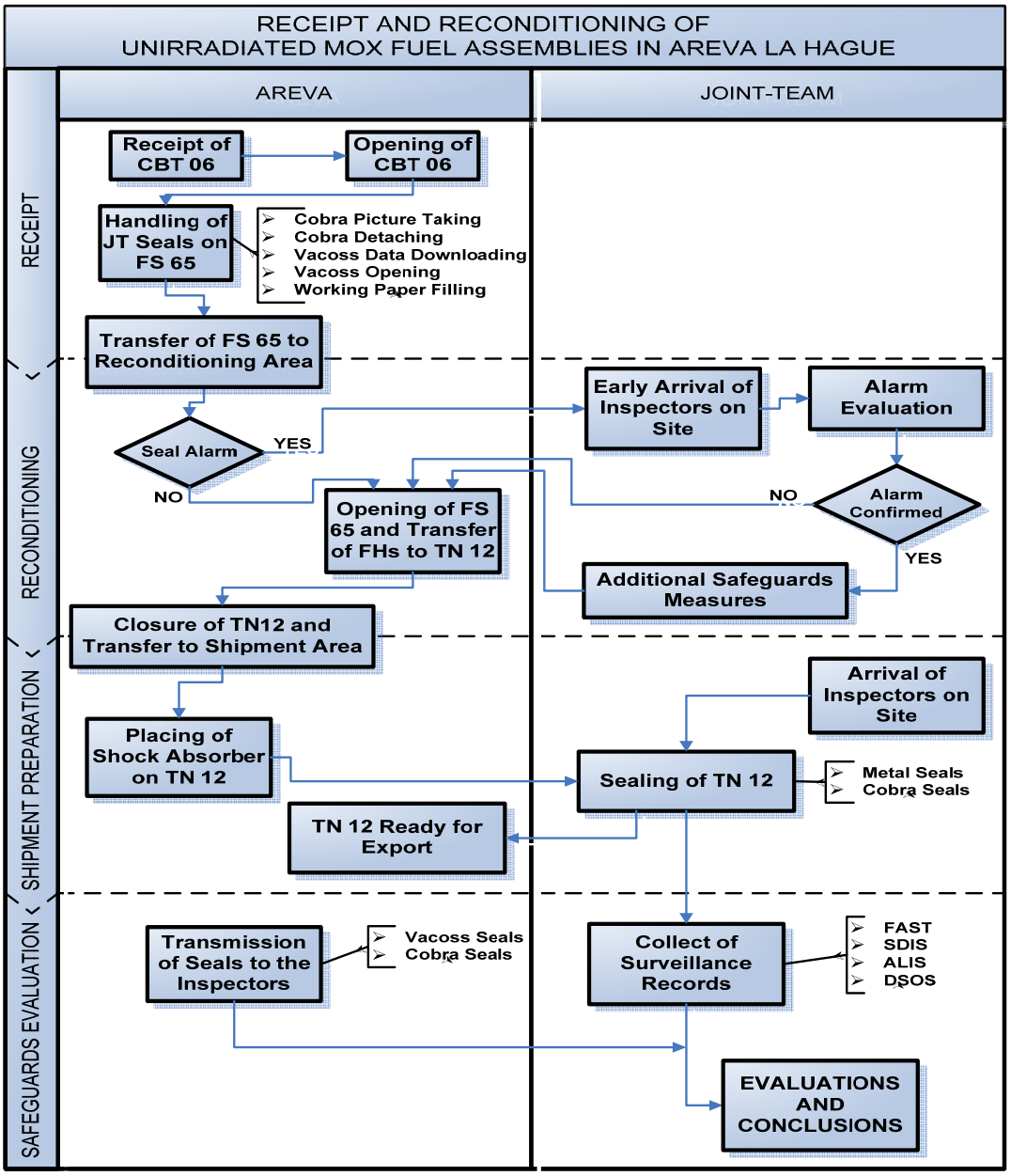

Figure 1: Inspection Scheme for the Receipt and Reconditioning of fresh MOX Fuel Assemblies in AREVA La Hague

If no alarm is transmitted to the inspectorates, arrangements are made with the operator so that Joint Team inspectors arrive on site when the TN 12 flask is ready for sealing. Inspectors seal the TN12, retrieve seals detached by the operator from the FS 65 transport containers, collect the relevant safeguards recordings and data in the plant and perform their evaluation.

3.4. Safeguards Equipment in use at La Hague

Because of the consequences in case of failures the implementation of a very robust, secure and redundant safeguards containment and surveillance system was felt absolutely necessary. In this respect, the surveillance equipment used in 1999-2000 for the 1st series of MOX reconditioning activities was not considered to fulfil these requirements and needed to be upgraded.

Considering the importance and extend of the upgrading and the relatively short preparation time, the quality of the project follow up and of the co-operation of all parties involved (Commission (inspection and technical units, financial services), operator, contractor and IAEA) was of utmost importance. A project co-ordinator from the EURATOM inspection unit was nominated and devoted a considerable amount of time to the project coordination.

3.4.1 Surveillance System Specificities

Particular attention was put to a complete redundancy of video systems from the source to the recording stations.

- All elements of the chain from every point of surveillance to the central recording were doubled up, including cameras, their power supply, signal transmission lines etc...
- Recording systems were also doubled up (one multi-camera FAST system and one multi-camera SDIS system), each system consisting of two recording units.
- Local recording systems (ALIS and DSOS) were also used to survey areas of particular interest or importance.
- Infra-red projectors were installed combined with high sensitivity CCD cameras to cover light failures.

3.4.2 Power supply

Particular emphasis was put on the power supply of safeguards video equipment. The two surveillance systems are fed through two separate power supply circuits. The SDIS system is connected to the Commission's power supply, the FAST system is directly connected to the AREVA secure power supply to mitigate the consequences of a single power supply failure.
Both SDIS and FAST recording units are also coupled to small uninterruptible power supply units (UPS) installed inside the recording cupboards in order to cover short power cuts.

3.4.3. Enhanced maintenance

All equipment has been incorporated in the maintenance contract already in place on site. In addition, during periods of shipment activities, the operator is responsible to control once every 24 hrs that safeguards surveillance systems are properly powered.

3.5. Feedback from the first reconditioning Activities at La Hague

From reconditioning activities that were carried out at La Hague during the second semester 2008, it can be concluded that the safeguards approach and implementation put in place at AREVA La Hague has proven to be robust and efficient.

No serious equipment failure was noted during the period and the delegation of seals handling by the operator worked in a very satisfactory manner.

With the organisation in place, safeguards activities concerning each TN 12 could be performed within 1.5 -2 days of inspection and the objective to optimise inspectors' presence on-site can be considered as fulfilled.

However, small organisational problems were noted that resulted from operational changes with short notice to the inspectorates.

4. Conclusion

The export of MOX fuel from MELOX to Japan has proven to be an example for effective and efficient safeguards implementation and cooperation that avoids a duplication of verifications by the inspectorates. The overall aim was to establish a verification scheme which is comprehensive, credible and allows both safeguards organisations to draw their independent conclusions, without creating an undue burden on the operator or excessive resource requirements on the inspectorates.

These arrangements are only possible because of the co-operation of all parties involved. The flexibility and co-operation shown by all stakeholders, especially on the operator's side, made it possible to attain the required safeguards goals without any undue repetition or doubling up of safeguards activities.

The delegation of safeguards activities, like the handling of seals by the operator at La Hague, requires adapted systems and dedicated training.

The arrangement that the IAEA gets the relevant subset of safeguards data collected by Euratom and carries out additional QA checks has proven to be a very efficient tool to avoid a duplication of safeguards activities and to limit the impact on production.

The networked nature of the safeguards instrumentation at MELOX allows for central data acquisition and simplifies the sharing of relevant data amongst the safeguards authorities. However, the evaluation of safeguards data at the installation is time consuming and often done under tight time constraints. The use of remote data transmission to the respective headquarters, as already in use at resources and to improve the efficiency of safeguards activities further.

Successful safeguards implementation is often a question of commitment and cooperation of all stakeholders and the MOX exports to Japan have proven to be a good example for further cooperation.

Feasibility of Using RFID in the Material Accountancy and Safeguards Verification at the Nuclear Fuel Cycle Facilities

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Abstract

New Safeguards concepts were studied to improve material accountancy and inspection efficiency to reduce both inspector and operator workloads for international safeguards. These concepts are based on new and innovative tracking and traceability technologies, such as tracking nuclear materials using Radio Frequency Identification (RFID), tracing nuclear materials using the isotopic composition of minor actinides, and tracking nuclear materials by monitoring plant operational parameters.
In the study, feasibility of using RFID in material accountancy and safeguards verification at the nuclear fuel cycle facilities was analyzed. Investigation on effectiveness in the safeguards, contributions to efficiency and cost-benefits of the RFID were evaluated and primary study on safeguards concepts using innovative tracking and traceability technologies for entire nuclear fuel cycle facilities were performed.
Testing to evaluate basic characteristics of commercially available RFID tags, such as readable distance, directional characteristics were performed. Irradiation testing of RFID tag by radioactive source and at research reactor to evaluate radiation effects has been performed.
Feasibility of using RFID in the actual condition was analyzed by small scale field test at JAEA Ningyo-Toge using selected RFID tags attached to UF6 cylinders and uranium scrap drums.

Keywords: safeguards concepts, RFID, tracking, radiation effects, field test

1. Introduction

New safeguards concepts focused on nuclear fuel cycle in the state as a whole based on new and innovative tracking and traceability technologies have been studied to improve material accountancy and inspection efficiency and reduce both inspector and operator workloads. This concept focus on new method such as tracking nuclear materials using Radio Frequency Identification (RFID), tracing nuclear materials using the isotopic composition of minor actinides, and tracking nuclear materials by monitoring plant operational parameters.

In the previous study, investigation of the RFID on effectiveness in the safeguards, contributions to efficiency and cost-benefits were evaluated and a primary study on safeguards concept using innovative tracking and traceability technologies, such as “near real time nuclear material tracking and locator in nuclear fuel cycle concept” was proposed and analyzed (see Figure 1) [1].

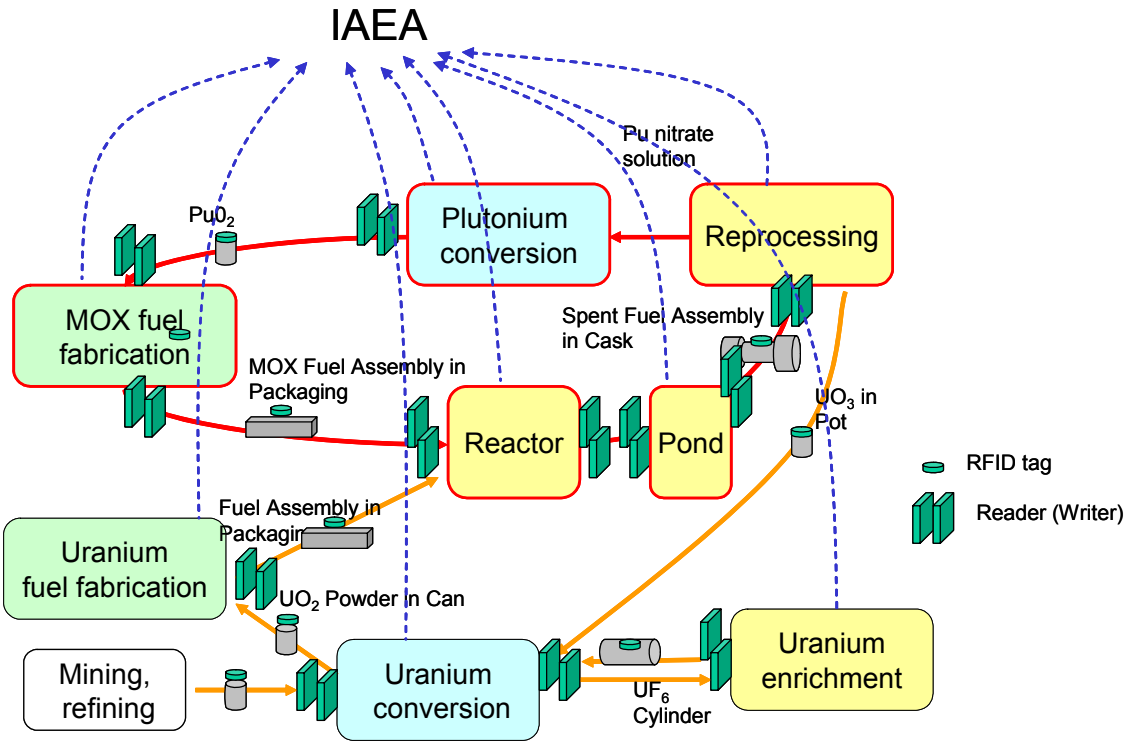

Figure 1. Image of "Near Real Time Nuclear Material Tracking and Locator in Nuclear Fuel Cycle".

Further study on the feasibility of RFID in the material accountancy and safeguards has been performed. In this study, (1) evaluation of basic characteristics of commercially available RFID tags and reader/writers to obtain basic data to select suitable RFID tags, (2) investigation of radiation effects by irradiation testing of RFID tag at Co-60 Irradiation Facility at JAEA Takasaki and JRR-3 research reactor at JAEA Tokai, (3) field testing of RFID tags at UF6 cylinder storage area of Ningyo-toge Uranium Enrichment Plant for the evaluation under real usage environment, (4) design and testing of prototype RFID tag with tamper-proof function for uranium drums have been performed.

2. Basic Characteristics of RFID Tags

Testing to evaluate basic characteristics of commercially available RFID tags and reader/writers to obtain basic data to select suitable RFID tags for the further feasibility study was performed.

Eleven types of RFID tags with different frequencies and different type of memory were selected for testing, taking into account broader application of RFID in safeguards field under wide-variety of usage environment including metal surface, high radiation and other. RFID tags selected for the testing are listed in Table 1.

Table 1. List of RFID tags selected for testing on basic characteristics.

No.	Type	Frequency	Memory
1	I-CODE SLI	13.56MHz	EEPROM
2	mifare	13.56MHz	EEPROM
3	HF FRAM	13.56MHz	FRAM
4	HiTag	125KHz	EEPROM
5	μ-Chip	2.45GHz	Fuse
6	μ-Chip Hibiki	UHF	EEPROM
7	UHF Omron	UHF	EEPROM
8	UHF Mitsubishi	UHF	EEPROM
9	UHF Fujitsu FRAM	UHF	FRAM
10	Secure Tag	426MHz	Flash
11	Active Tag	315MHz	

Testing to evaluate the readable distance between tag and reader, difference of the object material, such as wood and metal, directional characteristics, effect of radio wave absorbent and others were performed.

Directional characteristics of I-CODE SLI Circle φ20mm on wood and metal are shown in Figure 2. The shape of antenna and object material also contributes to the directional characteristics.

Figure 2. Directional characteristics on wood and metal (I-CODE SLI Circle φ20mm).

From the result of testing readable distance between tags and several types of readers, read/writable distance depends upon the size of antenna for in the tag for I-CODE SLI, on the other hand no dependency of the size of antenna for UHF type tags. Maximum read/writable distances are summarized in table 2.

Heat resistance was tested by soaking tags in 200 degrees Celsius silicon oil for 30minutues and liquid nitrogen for 30minutes. Water-proof was also tested by soaking tags into room temperature water for 30minutes. Results of the heat resistance and water proof testing are summarized in table 2.

Table 2. Readable distance of selected RFID tags.

No.	Type	Maximum Readable Distance	On Drum and 48Y Cylinder	Heat-resistance 200 degree Celcius	Liquid nitrogen	Water-proof
1	I-CODE SLI	290mm	20mm	OK	OK	OK
2	mifare	30mm				
3	HF FRAM	85mm		OK	OK	OK
4	HiTag	55mm	40mm			
5	μ-Chip	70mm	60mm			
6	μ-Chip Hibiki	5m	3.3m	Partially fail	Partially fail	OK
7	UHFOmron	10m	4.5m			
8	UHFMitsubishi	9.5m	4.5m			
9	UHF Fujitsu FRAM					
10	Secure Tag	>50m	>50m			
11	Active Tag	>50m	>50m			

In case the tags were attached to the metal surface such as drums and 48Y cylinder, the maximum readable distance was reduced. The effect of radio wave absorbents were also tested, and readable distances were increased when the radio wave absorbent was placed between a metal surface and tag for tags not designed for metal surface usage, and were not increased for the tags designed for metal surface usage.

3. Irradiation Testing of RFID Tag

In general, damage of RFID tags by radiation results in losing data in the memory at first. Therefore investigation of radiation effect was perform using several type of RFID tags with different ways of memory writing, such as EEPROM (Electrically Erasable Programmable Read Only Memory), flash memory, FRAM (Ferroelectric Random Access Memory) and fuse type memory.

Gamma ray irradiation testing was performed at several irradiation cells in the Co-60 Irradiation Facility at JAEA Takasaki (see Figure 3) for the total-dose evaluation testing.
Longer term irradiation testing was performed for the tags with higher radiation resistance.

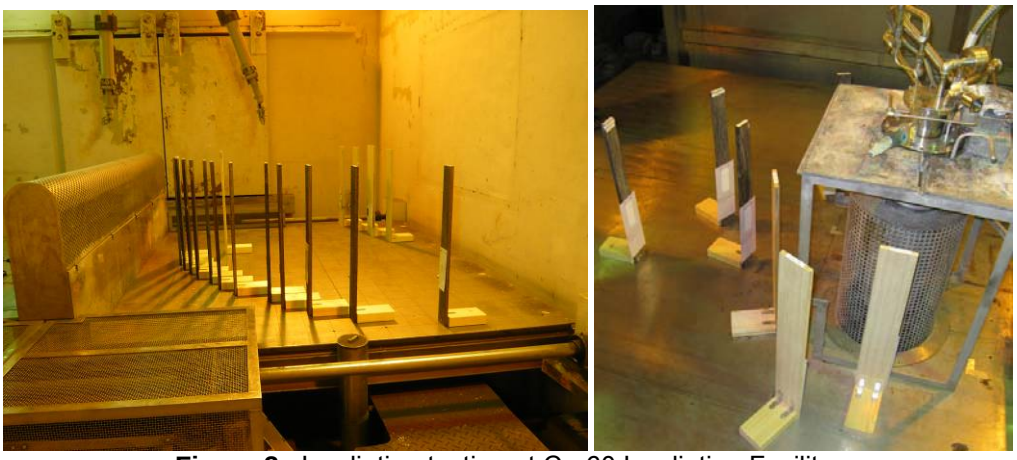

Figure 3. Irradiation testing at Co-60 Irradiation Facility.

9 types of RFID tags were irradiated for 100-1,680,000 Gy. The results of the gamma ray irradiation testing are summarized in Table 3.

Irradiation by neutron was tested at JRR-3 research reactor of JAEA Tokai (see figure 4). Selected tags were placed in irradiation capsules (rabbits) and transferred to JRR-3 core by pneumatic tube. The tags were irradiated for 20-60 seconds under $5.2X10^{13}n/cm^2sec$ neutron flux. Results of the neutron irradiation are summarized in Table 3.

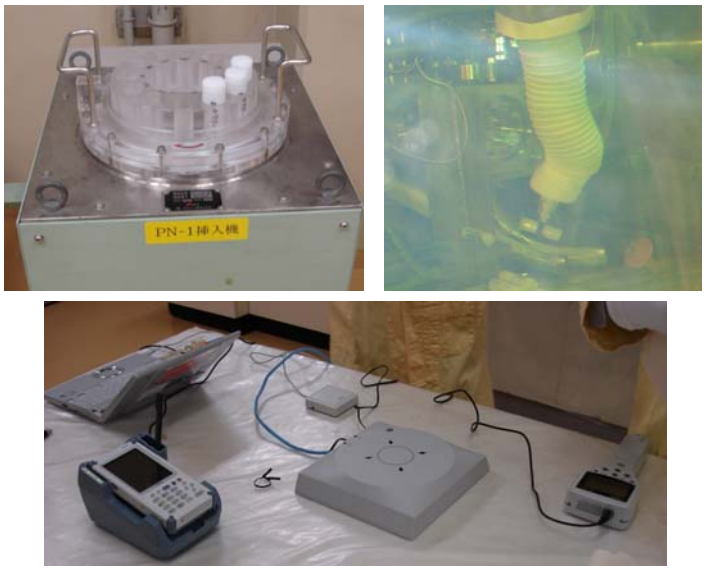

Figure 4. Irradiation testing at JRR-3. (Upper right) Irradiation capsules and pneumatic transfer system. (Upper left) Hot cell to handle irradiated capsules. (Lower) RFID readers.

The radiation resistance threshold obtained from gamma ray testing shows 250-300Gy in reading for I-CODE SLI. The threshold of HR FRAM andμ -Chip using FRAM and fuse type memory shows high radiation resistance.
From the testing, different radiation effects observed in reading and writing of memory. Different radiation damage was also observed in sectors with and without data in the memory. Recovery of memory's read and write function was observed in number of damaged RFID several hours after the irradiation.

Table 3. Summary of irradiation testing.

No.	Type	Memory	Gamma ray irradiation threshold	Neutron irradiation (20 sec)	Radiation dose after n irradiation (mSv/h)
1	I-CODE SLI	EEPROM	250 Gy	Fail	0.25
2	mifare	EEPROM	300Gy		
3	HF FRAM	FRAM	336,000Gy	OK	0.25
4	HiTag	EEPROM	250Gy		
5	μ-Chip	Fuse	>1,700,000Gy	OK	0.01
6	μ-Chip Hibiki	EEPROM	300-500Gy	Fail	0.1
7	UHFOmron	EEPROM	2,500Gy		
8	UHFMitsubishi	EEPROM	2,000Gy		
9	UHF Fujitsu FRAM	FRAM		Fail	0.01
10	Secure Tag	Flash			
11	Active Tag		2,000Gy		

Results from neutron irradiation shows almost same radiation effect as gamma ray irradiation results shown. Function of HF FRAM andμ -Chip were not be damaged under $1.0X10^{15}n/cm^2$ neutron flux.

Change of color was observed in the coating material of RFID after the long-term irradiation testing (see Figure 5). Zirconium system ceramic turns black and alumina system ceramic turns yellow by gamma ray irradiation. Change of color also observed after the neutron irradiation testing. Relatively high radiation doses were observed as shown in Table 3 after the irradiation. These are caused by radio-activation of materials in the tags by neutron irradiation. Figure 6 shows the gamma ray spectrometry of I-CODE LSI coated by ceramics after 60 seconds irradiation at $5.2X10^{13}n/cm^2sec$ neutron flux. Radioactive Cu, which is material of antenna, Zr, material of ceramics, and Na isotopes were identified.

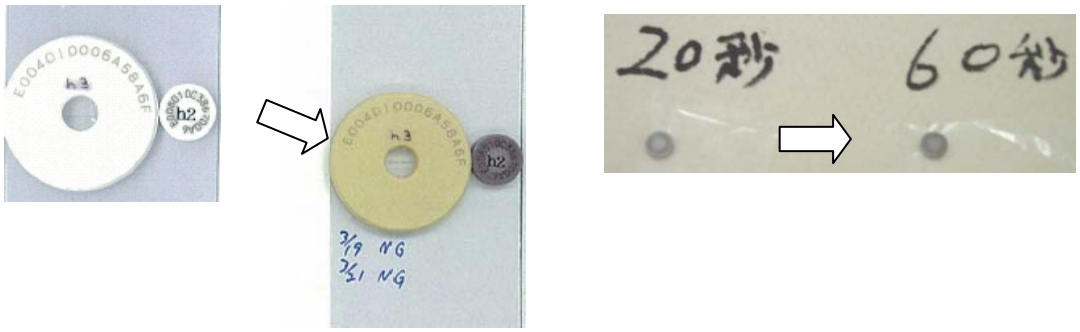

Figure 5. Change of color observed after irradiation testing. (Right) Color of ceramic tags before and after gamma ray irradiation (Left) color of ceramic tags after 20 and 60sec irradiation by neutron.

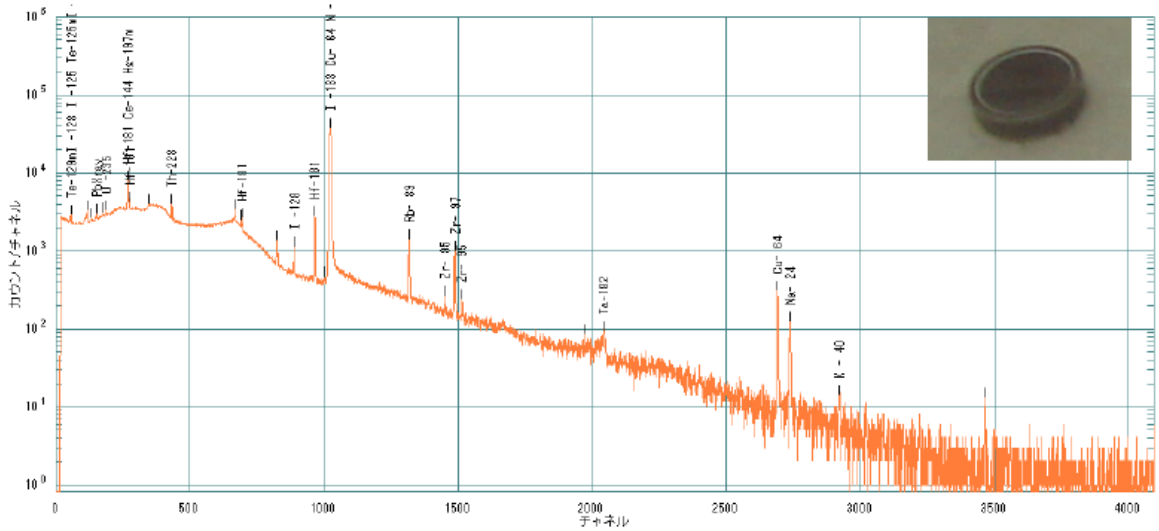

Figure 6. Gamma ray spectrometry of I-CODE LSI after 60 seconds irradiation

4. Field Testing at Ningyo-toge Uranium Enrichment Plant

Field testing of RFID tags using 48Y cylinders filled with depleted uranium was performed at the UF6 cylinder storage area of Ningyo-toge Uranium Enrichment Plant. The demonstration kit consisted of PDA type reader/writer and I-CODE SLI (13.56MHz) tags selected for the testing. Tags were attached to 36 of the 48Y cylinders (see Figure 7). The following information is necessary for material accountancy and safeguards are write into the tags; cylinder ID, batch number, material, tare weight, gross weight, net weight, element weight, enrichment, uranium concentration, date of measurement, IAEA seal number, JSGO seal number and location of cylinder. The date and time of reading the information in the tag is been recorded automatically.

Figure 7. Reading I-CODE SLI tag using PDA-type reader/writer on 48Y cylinder (right). Several RFID tags on 200 litter drum (left).

RFID tags can be read smoothly by a PDA-type reader and the time necessary for reading 36 cylinders was comparable with conventional item counting and visual tag confirmation (see Figure 6). It took about 2 minutes to read all 36 cylinders by PDA-type reader. It was almost the same time for conventional item counting and three to four times shorter than visual tag confirmation.

5. Design and Testing of Prototype RFID Tags with Tamper Proof Function

Prototype RFID tag with tamper proof function was developed.
The tag was design to attach to the clamp of lid of 200 litter drum by snap pin as shown in Figure 8. In case the snap pin is broken, function of tag will be lost. Therefore, after the snap pin is inserted to a hole of the clamp, removal of the tag can be detected by reading data in the tag.

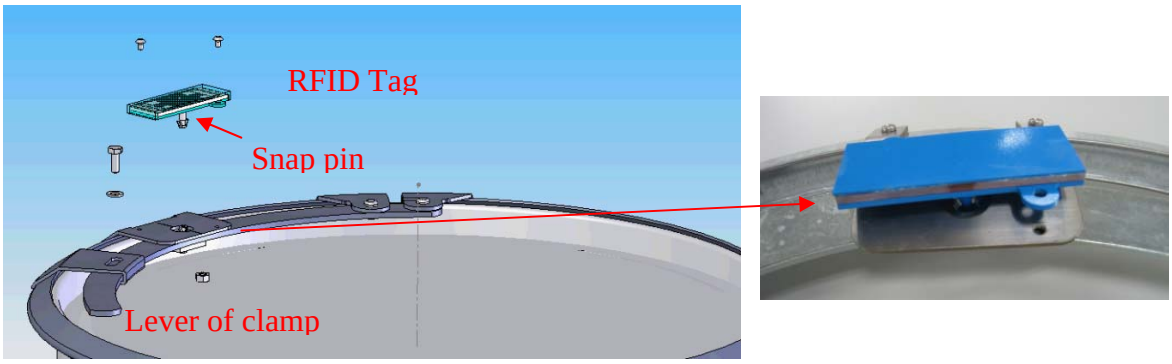

Figure 8. RFID tag with tamper proof function.

Uranium scrap drums monitoring system using FRID tags was studied. This system consists of FRID tags with tamper proof function, reader and writer, local computer and main computer connected with local computers by network (see Figure 9).
This system can be used as, C/S of nuclear material in drums, unattended ID check and item counting and supplement measure of seal verification for safeguards purpose and nuclear material control for facilities operational purpose.

Several testing of the prototype RFID tag was performed at JAEA Ningyo-toge to investigate feasibility of the monitoring system.

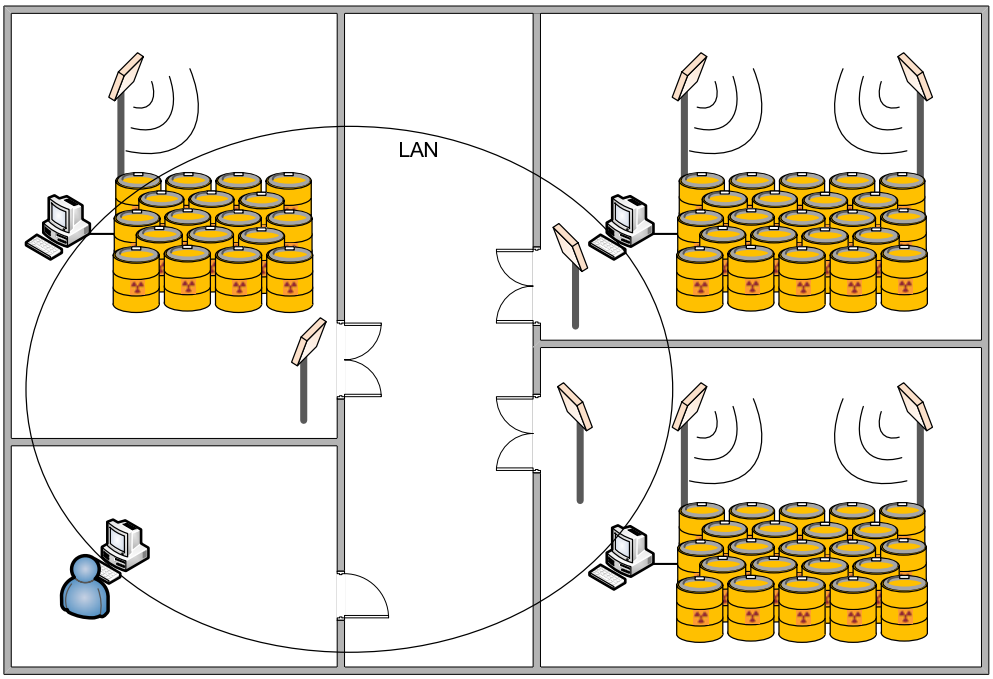

Figure 9. Image of uranium scrap drums monitoring system using RFID tags.

Conclusion

Results of the evaluation of basic characteristic and radiation resistance testing are summarized in Table 4. From the results, UHF type tags and some other tags shows good applicability for UF_6 cylinders the at uranium enrichment plant for routine material accountancy activities. The results show the use of these tags can be extended to other uranium handling facilities, such as 200 litter yellow cake dram at uranium conversion plant, UO_2 can and uranium fuel assembly packaging at uranium fuel fabrication plant.

Workability will be increased when using active tags. Higher cost and battery life are disadvantage of active tags.

From the irradiation testing, FRAM type RFID and μ-Chip show high radiation resistance. These tags can be use in the plutonium handling facility with relatively low radiation.
Recently, RFID coated by a wide variety of ceramics or other material are becoming available. Technical advancement in this area will extend the application of RFID tags to the higher radiation environment.

Table 4. Summary of basic RFID characteristics and radiation resistance testing

No.	Type	Readable distance	Radiation resistance	Cost	Remarks
1	I-CODE SLI	Relatively Short	Low	Cheap	
2	mifare	Relatively Short	Low	Cheap	
3	HF FRAM	Relatively Short	Medium	Relatively Cheap	
4	HiTag	Relatively Short	Low	Cheap	
5	μ-Chip	Relatively Short	High	Cheap	Read only
6	μ-Chip Hibiki	Relatively Long	Low	Cheap	With data security
7	UHF Omron	Relatively Long	Low	Relatively Cheap	
8	UHF Mitsubishi	Relatively Long	Low	(under development)	
9	UHF Fujitsu FRAM	Relatively Long	Medium	Relatively Cheap	
10	Secure Tag	Very long	Low	Expensive	
11	Active Tag	Very long	Low	Expensive	

There are several advantages of using these tags for current safeguards objectives, such as item counting, item identification, C/S and supplement measure of seal verification. To use the RFID tags for the safeguards objective, “tamper proof” and “authentication” are the functions that need to be applied to the RFID tags.
The prototype FRID demonstrated the possibility of adding tamper proof function to the FRID tag. RFID tags with data security function are being developed; for instance “the Secure IC Tag Project” in Japan. Under this project, memory of the tag can be divided to several user areas and read and write of data can be control by the password.

Study on uranium scrap drums monitoring system using RFID tags demonstrate the possibility of replacing surveillance cameras and seals currently applied for safeguards purpose with RFID tag system. And if this system can be expended to entire nuclear fuel cycle facilities, there is a possibility that inventory verification and flow verification activities at several facilities can be reduced by this system. The information provided to IAEA through this system will support their evaluation under State Level Integrated Safeguards.

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Developments in Remote Monitoring Data Processing at the IAEA

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Abstract:

The International Atomic Energy Agency (IAEA) has been using remote monitoring (RM) for safeguards since 1997. In the recent past, the availability of fast broadband Internet lines has enabled the expanded use of RM techniques. Over 60 unattended radiation monitoring systems and 100 surveillance systems are connected to the IAEA headquarters (HQ) or the regional offices. Remote monitoring offers great advantages: allowing for a reduction of inspection frequency at key nuclear facilities, lowering inspection costs for the IAEA and, limiting intrusiveness on facilities. The use of Virtual Private Network (VPN) has provided secure connections, with over 70 VPN tunnels worldwide. With this in mind, the IAEA must now transfer data in a more efficient and productive manner.

In order to optimise the process of transferring safeguards data, the IAEA has developed an in-house application which not only handles the transfer of over 3 GB of data daily, but inspects data files for valid content and authentication in an automated fashion. In addition to equipment State of Health (SoH) events being flagged and reported to technicians each morning, radiation data counts can be checked, missing images from surveillance cameras documented, and other data automatically authenticated. Furthermore, inspectors and operations management can use a web-based tool to view the real-time status of RM transfers and equipment health from facilities at their office desktops.

Keywords: Remote monitoring; nuclear facilities; safeguards data; Virtual Private Network; Latency

Introduction

The International Atomic Energy Agency (IAEA) has been using remote monitoring (RM) for safeguards since 1997. Remote monitoring is one of the cornerstones of modern safeguards and offers great advantages: allowing for a reduction of inspection frequency at key nuclear facilities, improving timeliness, lowering inspection costs for the IAEA, and limiting intrusiveness on facilities. In the beginning, only direct phone lines with analogue modems were available to transmit data from a relatively small number of systems. Since 2003, with advancements in the data communications technology, Asymmetric Digital Subscriber Lines (ADSL) have become widely available. ADSL lines are capable of transmitting data faster and more cost effective using the public Internet. Nowadays, the IAEA relies heavily on the public Internet using Virtual Private Networks (VPN) to ensure secure communications.

RM Network

The demand for remote data transmission services for safeguards increased sharply during the recent years and, in 2007, the IAEA established a dedicated Remote Monitoring Data Centre (RMDC) [1]. IAEA systems in the field collect safeguards data (e.g. pictures, radiation data, state of health (SoH)) from a device (camera, detector, electronic seal) and capture the acquired data in a local computer at the facility. Previously, such data were collected by inspectors during inspections and hand-carried to IAEA-Headquarters (HQ). RM data transmission techniques can extract all these data in regular intervals on a near real time basis from the field computer and send it to the RM data centre at IAEA-HQ. The transfer process has to be simple and robust; furthermore, the process has to download and organize the data into uniform data structures for evaluation and archiving at IAEA-HQ.

The major challenge for the RMDC is to provide field data to the inspectorate in a secure, reliable and timely manner. The RMDC must be able to retrieve archived data from the original data generator (e.g. camera) over periods of several months in case of failures in the communication lines or collection systems. In addition, the RMDC must be capable of responding to specific requests from the inspectorate (e.g. downloading data at short notice). Furthermore the RMDC should provide early warnings of malfunction in any field system.

The access to the collected RM data is controlled within the LAN environment to ensure that IAEA criteria concerning the confidentiality of data are strictly adhered to. All RM data is available to the relevant authorized inspector on a facility specific basis only.

The IAEA has developed its own processing tools for the RMDC to adapt to available communication and processing applications to its specific requirements such as safeguards confidentiality, enhanced reliability, and compatibility with other IAEA applications.

ADSL Connections

The performance of data transmission is characterized by two key parameters: bandwidth and latency. *Bandwidth* refers to the capacity of a line to transport data at a given rate and is measured in **bits**/s (**b**ps). Bandwidth is perceived as a measure of the speed of a connection. *Latency* refers to any delay in data transmission due to processing delays (e.g. arising as data pass through proxy servers, make network hops and observe protocol standards). The impact of latency on the transmission performance is further described in the next section.

While analogue modems have a limit of 56 kbps (realistically 33 kbps) in data transfer rates, standard ADSL connections are about 600 times faster and are very inexpensive (€30-100 per month for unlimited transfer volume); further enhancements in bandwidth can be expected in the future at no additional costs. The ADSL connections have only the disadvantage that they are optimized for download traffic, while the IAEA mostly requires the upload of data (from a facility DSL connection) to transfer data from the facility to Vienna. The download speed is presently around 2 Mbps, while the upload speed is usually less then 0.5 Mbps decreasing to 64 kbps. This is still significantly faster than normal phone lines and even faster than most satellite connections.

To ensure *secure* data transmission, the IAEA uses its own secure network on top of the public communication channels to encrypt and authenticate data. The use of VPN architecture - IPSec in tunnel mode – also hides the internal structure (IP-address scheme) from the public. Each connection has a separate tunnel endpoint with a dedicated hardware box (FIPS 140 certified) which manages all of the IPSec traffic of the connected computer without being noticed by the client.

Upload speed and Latency

Unfortunately, most existing unattended systems in use by the IAEA were not originally designed for remote data transfer. Furthermore, equipment providers often do not recommend modifying the original equipment control software for these purposes as this may impact upon the functionality and reliability of the equipment. Therefore providers discourage the installation of any smart software on the remote system, which might be needed to enhance data uploading; thus every modification has to be performed from the client side (IAEA-HQ). Only minor alterations, e.g. changes to IP-addresses, have been made on systems in the field.

Another problem area is the huge number of relatively small files (such as single image files). It is advantageous for the management of data processing to minimize the number of files at the receiving end and to combine and store all files generated in one day within a single file (archive file, similar to a zip file, but the entities are already compressed). Daily file sizes can vary between 1 k**byte** (1k**B**) for SoH or seals data and 40 MB per camera for video data. For simplicity, the files should be written to the archive file in chronological order, and the archive file should not contain duplicates. Therefore a software application was developed which collects the data chronology from the remote facility and writes it directly to the target archive.

Slow upload speeds were identified as the major inhibiting parameter for the RM network. The ADSL lines were tested (speedtest.net) before actual data transfer under field conditions took place and most fulfilled the provider's specification. When benchmarking the performance data of lines in actual field operation it was noted that data transfer was much slower then expected. The same results were obtained with other smart copy programs including **xcopy** and **robocopy**. The transfer rate often achieved only 20-25% of the expected performance.

To illustrate the differences observed, an upload speed of 256 kbps would result in an expected 26 kBps transmission rate (a rough calculation of throughput in Bps is estimated by dividing the bandwidth in bps by a factor of 10). In reality, only 5 kBps was achieved for transmissions between IAEA-HQs and the Tokyo regional office although the performance tool confirmed the specified higher speed. Further investigation revealed that the poor performance of the line was due solely to latency. Latency reflects the time needed for a sent packet of data to be received at its destination. It includes the time involved in encoding the packet for transmission, the time needed for that data to traverse the network equipment between the nodes, and the time taken in receiving and decoding the data. A minimum limit on latency is determined by the distance between communicating devices and the speed at which the signal propagates in the circuits (typically 70-95% of the speed of light).

A latency of 300 ms has been measured by a ping round trip between IAEA-HQ and the Tokyo regional office. The data signals from Vienna to Tokyo are usually routed via the US, travelling about 50000 km in a round trip. This already represents a significant part of the measured latency of at least 175 ms (50,000 km / (300,000 km / s * 0.95)). In this example, the packets are routed via more than 20 hops!

Data throughput is further reduced by delays arising from protocol driven acknowledgements in the transfer process. The worst case would be if each network packet (~ 1 kB) has to be acknowledged. In this case, the sender cannot send another packet before the current one is acknowledged. This reduces significantly the maximum throughput for 1 kB packets assuming infinite line speed and 300 ms for the latency, to only 3.3 kBps. The TCP/IP protocol - the protocol used for the public internet - applies a technique which is called "sliding window protocol", which reduces the number of acknowledgments (the sender can transmit up to 64 kB before waiting for an acknowledgment). With this optimization, the maximum throughput is increased to 213 kBps (64 kB / 0.3 s).

These limits do not apply to protocols where no acknowledgments are required (e.g. video/voice streaming). However, these protocols are not feasible for the remote transmission of SG data which must meet the highest standards of reliability. To implement reliable protocols with an absolute minimum of acknowledgments, additional software needs to be installed on the remote systems.

During the transfer process, Server Message Blocks (SMB: a part of the native file sharing protocol from Microsoft) are used to copy the data. The use of SMBs is the only alternative to installing third party components for a Windows operating system. By tracing the network traffic before the actual copy, a significant dialogue based preparation time was required for the transfer due to the authenticated inter-process communication. For each request, the system (at IAEA-HQ, Vienna) waits for the corresponding response from the remote computer (e.g. in Tokyo) and, because of the latency the system, never receives the answer in less than 300 ms. Actually, there are at minimum two requests (OPEN and STATUS) before the actual transfer starts (TRANSFER request) and one after the transfer (CLOSE), which means that the data is being received with a minimum delay of 1.2s.

It can be easily demonstrated that the latency is the limiting parameter for the transfer of small sized files because the initial delay is independent of the file size. The actual transfer time of an assumed 10kB file, transferred at a rate of 192 kbps (~ 20 kBps) and a latency time of 2 x 150 ms will take 1.7 s (3 x 0.3s + 0.3s + 0.5s) in total. This corresponds to an effective throughput of ~ 6 kBps (10 kB / 1.7 s). Assuming an infinite transfer speed with the same latency, would only increase the throughput to ~ 8.3 kBps (10 kB / 1.2 s).

File size in kB	Total Transfer Time	Throughput in kBps
1	1.2 s	< 1
10	1.2 s + 0.5 s	6
100	1.2 s + 5.0 s + 0.3 s	15
1000	1.2 s + 50 s + 16 * 0.3 s	18
10000	1.2 s + 500 s + 160 * 0.3 s	18

Table 1a: Throughput for different file sizes, a latency of 300 ms and a line speed of **192 kbp**

File size in kB	Total Transfer Time	Throughput in kBps
1	1.2 s	< 1
10	1.2 s + 0.1 s	8
100	1.2 s + 0.5 s + 0.3 s	50
1000	1.2 s + 5 s + 16 * 0.3 s	91
10000	1.2 s + 50 s + 160 * 0.3 s	101

Table 1b: Throughput for different file sizes, a latency of 300 ms and a line speed of **2 Mbps**

This means that increasing the line speed for the transfer of small size files (<10 kB) does not result in a significant benefit. On the other hand there is usually no inexpensive way to reduce the latency time for a given communication pair unless the number of nodes can be significantly reduced.

Multitasking

With such poor actual performance as described above, the IAEA would not be able to retrieve all needed data in a timely fashion. A software application ("DCCopy") has been developed to speed up the transfer. This tool uses XML initialization files and organizes file transfers in multitasking mode (parallel operation) to keep the line busy during latency. However, network resources and the operating system limit the number of parallel steps for the copy task. DCCopy performs seven concurrent transfers at any time. This is done in a quite simple way, as described below.

If there are less then seven transfers in progress, the system initializes the current transfer without waiting for the end (background process), instead it queues the necessary information to control and retrieve the returned data from the background task. After a short time there are seven tasks running in parallel. Before starting the next (8^{th}) task, the system has to wait for at least one task to finish. Because the data should be read in chronological order, the system waits for the oldest task (the first in the queue) to complete. It is unlikely that the background tasks in the queue (1^{st}-7^{th}) do not finish in the same order as they have been started, but this could happen. This is not a problem – if another task finishes first, the system still waits for the oldest task; which means that at this time only six tasks are active. When the first task finishes, it is removed from the queue and the system checks whether the next task (now the first in the queue) has finished (and so on until a busy task is identified or in the worst case the queue is empty which means the oldest task finishes last). If more than one task is removed from the queue, the next new task detects this situation and starts the transfer without any blocking (waiting) and after a short time up to seven tasks are again in process. Therefore, the line is kept busy at all times with background tasks and data is always copied in sequential order. With this technique it is possible to achieve transfer rates which are close to those expected limited solely by the transmission line specifications, and the archive is written sequentially without the need for an additional sorting process. The observed acceleration achieved is between a factor three and four. This software tool enables the retrieval of all SG relevant data from a facility in a reasonable time. A similar strategy also works for the transfer of larger files, but the achieved enhancement is *only* 30-40%. In the case of PSTN (phone lines) the same mechanism is applied, although the number of concurrent transfers is only two (3-4 kBps transfer rate) improving the speed by a factor of two.

Indexing

The introduction of a daily archive file (“dayfile”) minimizes the number of files, e.g. individual video pictures, to be uploaded and thereby significantly enhances the transmission performance. The dayfile consists of a file-header and a series of image-headers and image-data. For the transfer it is mandatory to know which pictures have already been copied to IAEA-HQ. This can be achieved by an index system which efficiently retrieves the information about the content of the archive. In this case only an index entry of a few bytes (8 B) is needed for each file in the archive. A maximum of 5000 pictures can be stored in a dayfile which requires a moderate index size of about 40 kB (the file size of the dayfile containing 5000 pictures is about 50 to 100 MB). The index file is obviously much faster to read than the entire day file. The index file is stored in an alternate data stream of the original file. The alternate data stream (also known as file system fork) offers the option to store additional data under the same file name. Currently there are only a few applications which use alternate data streams, but it is the ideal location for storing a file index. To protect a file against changes without corresponding index changes some vital information from the data file is stored in the index file. This ensures that if the data file is changed, the index becomes invalid and thus it will be noticed immediately.

Each RM system requires an exact knowledge of the state of the data in Vienna. To obtain this information the system has to check all local data; this could require reading several GB of data or traversing several huge directories, which require significant time (even when the process and the data are on the same machine). The smart index does not rely on a database and the processing time can be reduced dramatically (50 to 100 times faster). This software also simplifies and speeds up data transmission within a facility where a non RM approach is applied, e.g. due to lack of State permission for RM data transmission or a lack of regional communication infrastructure. It further supports the integration of SG data from various distributed collect stations to a central computer. There is no need for building a complex environment (such as a SQL Server) in the field, which would also require ongoing maintenance. In general complexity should be limited to servers in IAEA-HQ providing simple solutions for systems in the field.

SoH Database

All RM data is copied to the servers of the RMDC, and after pre-processing, fed by the downloading software to a database (SQL Server). This SQL Server is located in a different network at IAEA-HQ, is separated by a firewall from the RMDC, and holds all transfer status and SoH information. This means that each transfer file is processed only once at the RMDC. The remote data transmission from the field will also be performed when the SQL-server is down. In this case the database cannot immediately be populated, however, when the SQL-Server connection is later restored, the system recovers the missing database records automatically (self-healing) - usually there is no need to run the manual counting which is also available. Therefore the data in the database always reflect the actual status of the transferred file.

Web-based Monitoring Tools

The described software tool handles the transfer of over 3 GB of data daily and inspects data files for valid content and authentication in an automated fashion. Equipment SoH events are flagged and reported to technicians each morning. This application is now a `backbone` application for the RM data centre and additional web-based user interfaces have been developed which allow inspectors and operations managers to view from their office the real-time status of RM transfers and equipment SoH installed in “their” facilities. If the system detects a serious problem, it provides log entries within the daily emails generated to its subscribers. A website shows the real time state of the transfers. Whenever the download software contacts a facility computer it reads the time and the time zone from the remote computer and stores the difference to the local time in the database. This recorded time shift provides important information for systems which are not synchronized in real time.

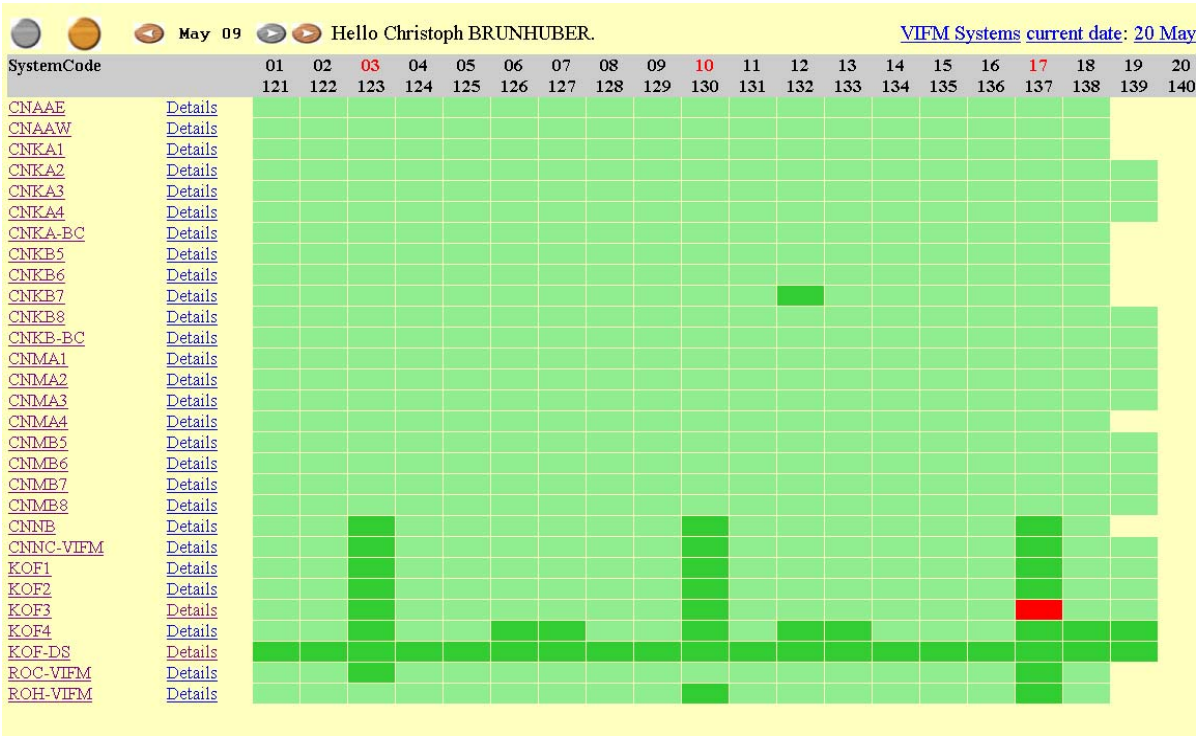

Fig.1: Status of RM file transfers

Fig.1 shows a typical screen shot from the transfer status monitor for a number of facilities using the same category of system. The ability to visualize the current status of RM transfers is a powerful tool to quickly diagnose the overall status of the connected devices. Each green box represents a successful upload for a calendar day. In the cases of malfunctions in a remote system (e.g. a loss in connectivity due to a defective collect computer) the colour would change to red to alert the user.

Achievements and Future Developments

Today, the IAEA has more than 170 unattended systems, including C/S systems, connected to the RM network. The network is connected to various locations in about 20 countries all over the world, transmitting measurement and C/S data and/or SoH data. Presently, the IAEA receives 3-4 GB/day of safeguards relevant data. The development and implementation of the described solution required approximately seven person-months. The HQ system has been fully integrated with the overall infrastructure of the RM network. Since its installation in 2008, considerable experience has been gained and about 2 TB of data have been remotely transferred. It is estimated that these tools have significantly reduced the monthly operating costs of the global RM network by a factor of three to four.

SoH events are being flagged and reported to technicians each morning and inspectors and operations management can use a web-based tool to view at their office desktops the real-time status of RM transfers and equipment SoH from facilities.

It is to be noted that the applications described are generic tools which would also support data transmission via satellite.

Future developments in RM data processing will aim to improve the quality control of transmitted data. Software tools are being considered for the detection of missing scenes, black frames and gaps in video and measurement data. The data could be automatically authenticated. Presently such activities, known as technical review, are performed separately out by safeguards inspectors prior to actual data evaluation. The envisaged automated technical review would significantly facilitate and shorten the overall data evaluation process.

Conclusions

Remote monitoring has become an indispensable tool for the implementation of safeguards approaches. Processing tools for RM data have been successfully developed and are routinely used in operating a large scale RM network and data centre. The software tools developed significantly increase the transfer speed of RM data for all file sizes and are generic regardless of the transmission medium. Without these custom tools, it would be impossible to maintain the efficiency of transfer required to cope with an ever-increasing amount of RM data. The IAEA cannot rely on commercially available utilities to manage this data transmission. Further development of the dedicated RM infrastructure is foreseen to ensure efficient, reliable and secure transfer of safeguards data from the field to IAEA-HQ including monitoring SoH of connected systems on a near real time basis. Moderate development efforts are required to enhance the automated technical review of RM data, which could result in yet savings of an inspector's time in performing data evaluation.

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A Concept for a World-wide System of Identification of UF_6 Cylinders

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Abstract

Standardised uranium hexafluoride (UF_6) cylinders are used at over 40 industrial facilities world-wide for conversion, enrichment and fuel fabrication. These cylinders are used for processing and storage, and over 50,000 tU in the form of UF_6 is transported each year in these cylinders. Although each cylinder is manufactured to an international standard that calls for a nameplate with the manufacturer's identification number and the owner's serial number engraved on it, these can be quite small and difficult to read. Therefore, cylinder handlers have used a wide variety of cylinder numbering systems, and many different methods are employed to record and read the number on a cylinder. Recognising that each facility seems to use a different identity number, a cylinder can have several different numbers recorded on it, by means of metal plates, sticky labels, paint or even marker pen as it travels among facilities around the world.

It would be beneficial to industry to have an international standard for numbering of cylinders, a modern method of recording the identity number on a cylinder, and ideally an automated method of reading of the cylinder number. This paper proposes a programme to achieve these aims, both for newly manufactured cylinders and for existing stocks.

If successful, there would be considerable benefit for operators in areas of transportation, logistics, administration, and nuclear materials accountancy and control.

Having an identification method that could be independently used by national authorities and international inspectorates could greatly increase the efficiency and effectiveness of safeguards verification as well. An improved verification system would provide increased confidence in matching reported shipments and receipts, quicker and more reliable verification of declared cylinder inventories, and an improved capability for safeguards inspectors to assure that no undeclared cylinders are present at a facility.

Keywords: *UF_6; cylinder; identification, ID*

1. Introduction

Standardized uranium hexafluoride (UF_6) cylinders are used at over 40 industrial facilities world-wide for conversion, enrichment and fuel fabrication. These cylinders are used for processing, transportation and storage, and over 50,000 tU in the form of UF_6 is handled each year in these cylinders. The identification (ID) number of a cylinder is recorded on it in a very wide variety of ways. This increases the workload for plant operators and safeguards inspectors in reading cylinder ID's and can lead to errors by misidentification.

Note that where this paper refers to a "UF_6 cylinder", what is meant is a 30 inch diameter cylinder (used for transport and storage of low enriched uranium at up to 5% U235 assay) or a 48 inch diameter cylinder (normally used for transport and storage of UF_6 at up to 1% U235 assay). Smaller cylinders (used for samples or for highly enriched uranium) are not considered in this paper: whilst the arguments about identification also apply to small cylinders, the total quantity of UF_6 being shipped in them is low. Improvements to ID systems for small UF_6 cylinders, or to other items such as overpacks, can be dealt with later.

2. Current methods used to identify UF_6 cylinders

Currently, operators use a variety of methods to record ID numbers on cylinders. Most cylinders have three numbers engraved on the nameplate (which is usually made of stainless steel and welded onto the curved end of a cylinder, on the end with the valve):

- **Manufacturer's serial number**. This information is mandatory on the nameplate, according to the UF_6 cylinder standard ANSI N14.1. The full name is usually abbreviated, for example to "mfg. serial no". On some older cylinders, no manufacturer's serial number is recorded.
- **Owner's serial number**. This information is also mandatory on the nameplate, according to ANSI N14.1. It is often referred to by other names, such as "unique no.", "cyl no", or "owner no." The number is usually preceded by the name of the owner. But the owner name recorded is often the original owner: if the cylinder has been sold on, then the current owner is often not shown. Occasionally, the name of the owner is not recorded at all.
- **National board registration number**. This information is also mandatory on the nameplate, according to ANSI N14.1. Other pressure vessel body numbers may appear on ISO 7195 cylinders.

The ID numbers are recorded on a cylinder in a wide variety of methods:

- By engraving on the nameplate (which is usually made of stainless steel and is welded onto the curved end of a cylinder, below the valve). According to ANSI N14.1, the digits or letters must be a minimum of 5/32 inch (4 mm) high on a 30B cylinder and ¼ inch (6mm) high on a 48 inch cylinder. According to the UF_6 cylinder standard ISO 7195, the digits or letters must be a minimum of 8 mm (5/16 inch) high on both types of cylinder.
- Printed on a label stuck on the cylinder. Usually on the valve end, sometimes on the plug end, sometimes on the side or top.
- Painted on the cylinder, usually with stencils.
- Written on by hand using a large felt-tipped pen.
- Written on by hand in chalk.

In addition to the mandatory nameplate, cylinders can sometimes have a second nameplate, anything between one and five ID labels, numbers painted on, and written in ink or chalk as well! And some of these labels can also have bar codes; sometimes two or three different bar codes are on the same cylinder. It does seem that each new user adds his own ID marking.

The ID numbers recorded can take a wide variety of forms, using a range of alphanumeric characters. Examples of actual ID numbers on different cylinders are:

1629 – 229C;
TNF.130;
1629/000426;
F3112.5;
62991;
1125/CA 0050;
0184;
28292-16;
683/029;
572122;
5.55018;
20893-104;
20.662-002;
4007224

3. Problems arising from current methods used to identify UF_6 cylinders

Because of the plethora of methods used to record ID numbers, the following problems can arise:

- There can be confusion over which of the different numbers should be used. Should a nuclear facility primarily use the owner's serial number or should it use the manufacturer's serial number? Or should it use a dummy, internal number instead?
- Sometimes care must be taken to avoid misidentifying a cylinder. For example, a cylinder could have the owner number marked on it in two or more slightly different ways - and then which of these should be recorded on the computer system, and on the delivery paperwork? Indeed, sometimes the number on the delivery paperwork doesn't precisely match any number shown on the cylinder.
- Sometimes care must be taken to avoid the same cylinder being recorded more than once in a database, with slightly different ID numbers.
- Each company has to decide how to interpret non-alphanumeric characters. For example, should a space be ignored? And is a decimal point seen on the nameplate to be recorded, or is it just a speck of dirt?
- If a site wishes to use bar code readers for identifying cylinders, then it must affix its own bar code labels. It is never possible to read and make use of bar codes on labels which other companies have applied, because each company uses a different barcode number - or even a different barcode technology as well.
- Many of the numbers are too small to see clearly – particularly those on the nameplate. And the contrast can be poor, particularly if the cylinder is old and the metal has become tarnished or is dirty.
- Because numbers are mainly read by eye, there can be transcription errors (so cylinder 20893-104 might be recorded as 20983-104. If the latter cylinder is also present at the same site, then this error might go unnoticed).
- Each company can record a different cylinder ID number (maybe only slightly different). When a cylinder tours the world, then the number recorded on the paperwork can be different for each site it visits. This can make it hard for operators to track cylinders and for international safeguards inspectorates such as Euratom and IAEA to match declared shipments and receipts at different facilities.
- When a cylinders changes owner, it is difficult to amend the owner shown on the nameplate. Sometimes it is impossible (e.g. because there is no space to do so), and this can case administrative problems later, as sites which handle these containers in years to come may not be aware of who the current owner is.

In practice, both operators and safeguards inspectors carry out their work with great diligence, and it is rare for the problems described above to lead to a serious problem of misidentification. Nevertheless, it would be far better if the potential problems did not exist.

4. Suggested criteria for a modern, efficient cylinder ID system

The following criteria are desirable for a modern, efficient cylinder ID system:

- There should be a single, unique, unchanging ID number on a cylinder.
- The ID number should be globally recognised and accepted.
- The ID number should be of a fixed, recognisable pattern.
- The number should be marked on the cylinder in large print, such that it is easily visible by eye from a reasonable distance away. (Then there would be no need to paint the number on a cylinder as well, or affix lots of labels. Such ad-hoc methods of marking ID numbers on cylinders should not be necessary, and this practice should be discouraged.)

- There should be agreed technologies for automated reading of the ID number, using compatible technologies that allow automated reading for accurate and consistent information flow. (This would speed up the acquisition of data, and would markedly reduce transcription errors.)
- The ID should be difficult to forge[1]. (This criterion suggests that it should be permanently affixed to the cylinder, maybe as part of the nameplate.)
- The ID should survive all of the environmental conditions to which UF_6 cylinders are subjected during their processing and transportation.
- There should be a consistent location for affixing the ID. (And maybe even multiple locations to ensure that the numbers are viewable.)

If one looks at the international tracking of individuals, these criteria are the same as have already been implemented for passports. One's passport is nowadays easily machine-read by immigration officials at ports of entry, in most countries of the world. One has no need to deface one's passport by writing on it the ID number by hand – indeed we think that such a practice would be frowned upon by immigration officials!

5. Proposed plan to develop and implement a world-wide ID system for cylinders

The proposed programme to achieve this aim comprises the following elements:

- Formation of an international working group, including representation from major nuclear operators, Euratom/IAEA and USDOE/NNSA;
- A review of current methods being used in industry for identifying cylinders;
- Development of a world-wide unique ID numbering system for cylinders;
- Development of a method for marking the ID number on a cylinder and development of technologies to automate reading the number; this may include field trials at one or more industrial facilities;
- Development of an agreed tamper-indicating methodology, to provide assurance that the ID has not been forged;
- International agreement on the ID system;
- Development of an approach for implementing this cylinder ID system for existing stocks;
- Investigation of how to incorporate the ID into the cylinder manufacturing process.
- Incorporation into the ANSI N14.1/ ISO 7195 UF_6 cylinder standard for manufacture of new cylinders.

[1] Ideally, the ID should also be difficult to duplicate. Otherwise, a theoretical scenario for an operator to hide the processing of undeclared uranium could involve using two identical cylinders with the same ID (one would contain the declared uranium, whilst the other would contain the undeclared uranium). We are not aware though that this has ever happened in practice. It would probably be difficult for inspectors to recognise such duplicates, and to meet such an aim is outside the scope of this paper. Nevertheless, successful implementation of the programme proposed would assist the development of a system to detect duplicate cylinder IDs in the longer term.

6. Benefits of a modern, efficient cylinder ID system

The following benefits could accrue if the proposed programme were successfully implemented:

- Near elimination of transcription errors.
- More rapid reading of ID numbers, with consequent saving in time and radiation dose, both to plant operators and safeguards inspectors.
- Consistency of paperwork, administration and reporting to authorities.
- Increased confidence in safeguards inspectorates' matching reported shipments and receipts.
- Quicker and less error-prone verification of declared cylinder inventories by safeguards inspectors.
- An improved capability for safeguards inspectors to assure that no undeclared UF_6 is present at a facility.
- Avoidance of the need for operators to use ad-hoc methods to mark ID numbers on cylinders, by means of printed labels, stencilled paint or handwritten chalk or marker pen.
- The possibility of an integrated track-and-trace system for cylinders.

These benefits would accrue over the whole lifetime of a cylinder.

7. Conclusions

Currently, operators use a variety of methods to record the ID number on cylinders used for storage and transport of UF_6. Identification errors are avoided only because of the skill and diligence of operators and inspectors. This paper proposes a programme to develop and implement a standard world-wide system for identification of UF_6 cylinders. If successful, there could be considerable benefits for operators and for safeguards inspectorates in areas of transportation, logistics, administration, nuclear materials accountancy and control, and international safeguards verification.

Note: This paper will be presented at the 31st Annual Meeting of the European Safeguards Research and Development Association (ESARDA); Vilnius, Lithuania; 26 to 28 May 2009.

SESSION 8

NATIONAL LEVEL AND INTEGRATED SAFEGUARDS

Sweden towards integrated safeguards

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Abstract

After Sweden, together with the other member states of the European Union and the European Commission, signed the Additional Protocol 1998, preparations have been ongoing for implementing Integrated Safeguards, IS.

Sweden had its needed revised legislation to fulfil the requirements of the Additional Protocol, AP, in force early 2000 and ratified the AP in May 2000 and on April 30, 2004 the AP entered into force in the European Union.

Since then a long process have been going on to prepare for and implement Integrated Safeguards in the EU states. For Sweden, the IAEA have drawn the necessary conclusions to start implementation. There will be a mixture of short notice random inspections and unannounced inspections. During 2008 discussions with the IAEA, the Commission, the State authority and operators have been performed to pave the road towards IS. The most difficult task have been the LEU fuel fabrication plant but also for the state authority to arrange so that it's inspectors can, with very short notice, get to the facilities.

At the same time there has been a reorganisation of the safeguards authority in Sweden which of course did not help the process.

This presentation will describe how we in Sweden have come to organise the implementation of IS on all levels including the communication ways with the IAEA and the European Commission.

Introduction

Sweden started it's nuclear ambitions, as many other states, during the 40ties. In fact only a couple of weeks after the Hiroshima bomb, a more structured research on the development of a Swedish option for use of nuclear, both civil and military was initiated. The idea was to use natural uranium as there are quite vast resources within Sweden.

Internationally the progress in the development of the nuclear industry was made possible by the "Atoms for Peace Program" initiated by President Eisenhower in 1953. This made it possible to have trade with nuclear material and equipment. But the other side of the coin was that you also needed control of these activities which was the embryo of the Non-proliferation regime.

The first research reactor in Sweden was started 1954 in a rock cavern under the Royal Institute of Technology, KTH in Stockholm and at the same time a research establishment was built 100 km south of Stockholm at Studsvik. This was the start of an intense nuclear development, research projects to realize the Swedish line were launched, prospecting for uranium was initiated, a uranium mine was started 1965, locations for possible reactor sites were identified and a location for a reprocessing plant was selected. At the same time the

Swedish National Defence Research Establishment had ongoing research on the use of a nuclear option for military purposes.

Both the military and civil progress went along side by side during the 50ties and 60ties. The first commercial reactor was started 1964 in the southern part of Stockholm producing both electricity and heat to one of the suburbs. But more and more resistance began against the military part of the nuclear program and by the end of the sixties when the Non-Proliferation Treaty, NPT was negotiated, Sweden was an active part and Sweden ratified the NPT 1970. At the same time Sweden abandoned its natural uranium program as the price for uranium internationally had become so low that it was cheaper to by enriched uranium from abroad then producing our own natural uranium. So the uranium mine was shut down, the military research was stopped and new types of light water reactors were constructed, the first one, a Swedish design BWR in Oskarshamn, went into operation in 1972.

In 1972 the first agreement with the IAEA was negotiated together with the US. It was a trilateral agreement which only covered US-obligated material. In 1975 a comprehensive safeguards agreement went into force according to the model agreement INFCIRC/153. At that time Sweden had already developed its own State System for Accountancy and Control and nuclear safeguards in Sweden was to be handled by the Swedish Nuclear Power Inspectorate that started 1974.

During the rest of the 70ies and the 80ies there was an expansion in the use nuclear energy but one important factor happened and that was after the Harrisburg incident. The use of nuclear was questioned and the Swedish government decided to hold a referendum 1980 on the future of nuclear use in Sweden. The result was that maximum 12 nuclear reactor was to be built, including those already in operation, and eventually also a decision was taken by the Parliament to face out nuclear by 2010. The last reactor began operation in 1985. Also a condition was set for using nuclear energy and that was to present a method for taking care of the spent fuel. During the 70ies the option of reprocessing was chosen and contracts were signed with both Sellafield and La Hague. But this changed and the option became direct disposal in the Swedish bedrock. For this purpose a common intermediate storage for spent fuel was build for all Swedish reactors and the storage started its operation in 1985 and site investigations were initiated to find a suitable location for a geological repository for the final storage.

The Strengthened Safeguards System, 93+2

Safeguards were operated smoothly and no major disturbances occurred until the Iraq war 1991 when the clandestine nuclear program of Iraq was revealed. This caused activities among those involved in safeguards matters. The director of safeguards division at the Swedish Nuclear Power Inspectorate, Mr Paul Ek, was at this time the chair of SAGSI and SAGSI initiated a lot of measures how to strengthen safeguards. This became the so called 93+2 Programme that was initiated 1993 with the aim to be finalized within two years. Many states were involved in different field trials to test elements to strengthen safeguards and so was also Sweden.

Field trials in Sweden under 93+2

Sweden became engaged in testing four elements to strengthen safeguards: a so called expanded declaration, increased and more timely information flow, unannounced inspections and environmental sampling. I will give a short description on these four elements.

The expanded declaration: There was a proposal to evaluate the use of a more detailed information on the state's nuclear program, both past, present and future. As Sweden had had a two line programme during the 50ties and 60ties it was a good exercise to describe the Swedish situation and we found that the description of the past activities was not easy to perform though it was found of value to be able to explain what remained from the early activities. The expanded declaration eventually turned out to become the declaration format under the Addition Protocol after quite a few improvements.

Sweden also volunteered to test a timelier and also increased information flow. As the Swedish SSAC requires the operators to roughly daily report inventory changes to the Swedish authority the authority always have up to date information on the nuclear material in the State. The agreement with the Agency for the field trial was to electronically submit weekly ICRs. In addition it was agreed, concerning the fuel fabrication plant, to every Friday submit information on the next week's planned production. This information together with the ICR information was then used by the Agency to plan unannounced inspections.

An unannounced inspection scheme was also tested involving the power reactors, the fuel fabrication plant and the research facility. The agreement was that the Agency inspectors would show up at the gate of the facility showing an inspection assignment to the guard. The operator should then immediately inform the authority who would send an inspector to the facility. The IAEA would have access to the facility within 2 hours and if the state inspector still had not arrived the operator would represent the state until the inspector arrived. In total there were 5 or 6 unannounced inspections performed during a one year trial.

The last element of strengthened safeguards to be tested in Sweden was the use of environmental sampling. This was done in the surroundings of three of the reactor sites, the research facility and the fuel fabrication plant. Samples were taken both from soil and the sea.

The Additional Protocol and EU

The IAEA Board of Governors approved the proposal for an additional protocol to the comprehensive safeguards agreements at its May meeting 1997. This was in fact during the ESARDA symposium in Montpellier and Mr Murakami from the IAEA got the honour to announce this to the participants of the symposium on the same day the decision was taken.

Now the process started to prepare for signing and ratifying the Protocol. As Sweden from January 1st, 1995 had joined the European Union the situation had changed for Sweden. Sweden was now party to INFCIRC/193 and the European commission was now the contact for the IAEA regarding the comprehensive safeguards agreement regarding the Swedish facilities. But the additional protocol also covers areas not involving nuclear material so quite intense discussions were held within the EU concerning implementing the AP. Eventually the decision was taken that the AP would enter into force in all EU-member states (including the

two NWS) at the same time. The AP was signed on September 27, 1998 with the ambition to have the AP ratified by the NPT review conference 2000.

There became intense discussion on the role of the different parties in implementing the obligations of the AP in EU. As there already in the Protocol was identified shared responsibilities between the Commission and the States there was an agreement that those states, who so wished, could ask the Commission to perform all obligations for that state concerning the AP. This was done through submission of a so called side-letter to the Commission. So there became a two way solution in EU, side-letter states and non-side-letter states.

In parallel the old safeguards regulation 3227/76 in EU was subject to revision to reflect the new situation with the AP and also to be updated to the situation in the new millennium. That process went on during 2002 until the end of 2004 with a lot of involvement from the European Commission, the Member States and the Atomic Questions Group of the European Council. The new regulation called Regulation (Euratom) No 302/2005 on the application of Euratom safeguards entered into force in March 2005.

Swedish preparations between signature and entry into force of the AP

As Sweden had been involved in the field trials there were already some pieces in place but one lesson learned was that we needed to know better about Sweden's past activities, so several research projects were initiated to make a historical review of "nuclear Sweden". This initiative was taken to be able to give answer to possible questions from the IAEA about what happened with the old nuclear programme and facilities. We also had to make a survey to be sure to get hold of all who were involved in production, import and export of nuclear related equipment and material. The work to identify those who were involved in nuclear related research was also a major task.

The Swedish Support Programme to IAEA safeguards had a task looking at the feasibility for the IAEA and also the States to use satellite imagery for checking site declarations. We choose two sites for this task, the reactor site at Oskarshamn and the research complex at Studsvik. For this satellite images were ordered and at the same time we also ordered images for the other reactor sites in Sweden for use to check the completeness of the declarations when they were received from the sites.

On the legal side we had to analyse what changes and additions to the existing legislation we had to perform to be able to ratify and comply with the AP. The result of that were additions in the existing law on nuclear activities to also to include an obligation to report on nuclear related research and nuclear related equipment. We also had to introduce a new law to be able to grant access to facilities not covered by the law on nuclear activities.

And finally we also had to inform the nuclear industry on the new obligations resulting from the AP. That was done through visits by the Authority to all who somehow had to submit information to the Authority to enable Sweden to be able to fulfil the obligations of the AP. The installations that were regarded to become a site were asked to fill in a site declaration and to submit a draft to the Authority. At this time we hade also got CDs with the software for the IAEA Protocol Reporter that also was handed over to the safeguards personnel at the facility so they could use that tool for making the declaration. To help the process both representatives from the IAEA and the Commission visited some of sites together with the

Swedish Authority to discuss and agree on the size of the site and how detailed the information should be.

Sweden decided to be a non-side letter state. This means that Sweden is responsible for articles 2a(i), 2a(iv), 2a(ix), 2a(x) and 2b(i). The Commission is responsible for 2a(v), 2a(vi) and 2a(vii). For the articles 2a(iii) and 2a(viii) there is a shared responsibility resulting in that Sweden prepares the declaration and submits it to the Commission who then, after checking the content, forwards it to the IAEA. The regulation 302/2005 has a requirement to nominate a so called "site responsible" for each site. Sweden decided to nominate its Authority as the site representative for all sites in Sweden meaning that the operators report on changes of the site to the Authority and then the information is compiled for all of Sweden before submission to the Commission and the IAEA.

Entry into force of the AP and the route to Integrated Safeguards

Sweden ratified the AP in May 2000 but the entry into force was on not until April 30, 2004. Once the AP entered into force the installations who some years earlier had sent their declarations where asked to update the information to reflect the situation by April 30. Eventually there were only 8 sites that became candidates for the declaration. For those that were not included, an attachment to the declaration gave motivations for the exclusion from a site declaration. Examples were small installations for which exemptions were asked, old closed down facilities without DIQ/BTC. Some sites were declared with a smaller site area then was discussed earlier and for those we described the buildings not included in the site in an attachment to the declaration. And finally to complete the whole picture we also attached all the research reports published on the nuclear history of Sweden.

Sweden decided to use the reporting tool CAPE, developed by the European Commission, instead of the Protocol reporter. By choosing CAPE we had to convert the information received from the sites in Protocol Reporter format to CAPE format. The converter tool was to rewrite the information in CAPE. The first declaration from Sweden, 2a(ixa), the export declaration was submitted on August 5th, 2004, while the site declarations and the waste declaration were sent to the Commission end of September and the rest of the declarations directly to the IAEA on October 20. For us the most complicated declaration was the research declarations as it was difficult both to identify the possible actors for reporting research and also which projects we would select to finely be declared. In total Sweden ended up with about 30 projects.

Now the administrative parts were done from the Swedish side and we just had to wait and see how the IAEA would react on the declaration.

For the updates of the AP-declaration the Authority informed the facilities that from 2005 they could do the updates through an ordinary letter and they we not obliged to use either Protocol Reporter or CAPE. It was easier for the Authority to handle plain text and manually input the information into the database as we had found that there was not a lot of information to be handled.

The first Complementary Access was with 2 hours notice and took place at the Studsvik research facility on March 16, 2005 in conjunction with the yearly PIV. The main issue to be discussed during this first CA was the Agency use of camera for visual observation. Sweden would not allow the Agency to use the camera and offered instead a camera supplied by the

operator. Then the Agency inspector received copies of the pictures taken after clearance by the facility security staff. The copies were received on a CD. This has then been the agreed procedure for use of cameras. This CA was then followed by another four during 2005, one of them with 24 hour notice to the closed down Barsebäck site.

In July 2005 the IAEA sent out a letter asking for clarifications on the initial declaration. In fact the letter was split into three parts according to the responsibilities defined in the additional protocol for the EU non-nuclear weapons states. In total there were 50 questions. It was interesting to note that when the AP-declaration was first submitted, Sweden attached information on the nuclear history but that had not included the uranium prospecting an mining activities during the 50ties and 60ties. And of course there were questions asked about those activities and what became of them. So lesson learned was for the historical review to start from the earliest point where the state starts its nuclear program.

In 2006 another letter asking clarification was received and that was linked to findings during one of last CAs during 2005. This was all complementary information that the IAEA looked for after the AP-declaration was first submitted until the broader conclusions were drawn for Sweden.

During 2007 and 2008 we noted there were more activity from the side of both the IAEA and the Commission. The two organisations discussed in both the HLLC and LLLC on how the cooperation had to be adapted to be able to cope with the approaches under the integrated safeguards regime. Sweden and Finland together had joint meetings with the IAEA and the Commission both 2007 and 2008 to discuss an integrated safeguards approach for Finland and Sweden. Finland reached an agreement on the approach early fall 2008 and IS was introduced in Finland October 15th, 2008.

For Sweden the overall agreement on the implementation of IS was finalized late fall but we still had to agree on the approach for the fuel fabrication plant. A joint meeting with the operator, IAEA, Commission and the new Swedish authority responsible for safeguards, the Swedish Radiation Safety Authority, SSM, was held in the beginning of November where the discussions on the IS approach for the fuel fabrication plant started from the generic PA-IS document for LEU fuel fabrication plants that had been approved by the HLLC. After a full day with very constructive discussions from all sides we managed to come to an agreed PA-IS for the Västerås facility. The only remaining issue was to have a functioning mailbox system which had to wait until the facility had finalized its ongoing upgrading of its safeguards accountancy system.

As we had the impression that the IAEA wanted to finalize the process of introducing IS in Sweden during 2008, the SSM made visits to Studsvik research facility and Clab, the intermediate storage for spent fuel, because these two facilities would be subject to unannounced inspections. During these visits procedures for granting access for the IAEA and for the contacts with SSM was discussed with both the safeguards and security staff.

Finally there was a short meeting between representatives from the IAEA and SSM in Vienna in conjunction with another meeting early December to confirm that all preparations for beginning IS were done from both sides. The letter confirming that IS would start in Sweden from January 15th, 2009 was then sent on December 19th to the Commission with a copy to SSM.

The integrated safeguards approach for Sweden

For Sweden the approach eventually turned out to be as follows:

For the 10 operating reactors there is a SNRI regime with 48 hours notice and at least 3 inspections in total for all 10 reactors. In conjunction with the PITs there are two inspections, a pre-PIV and a post-PIV with surveillance during the period when the core is open.

For the fuel fabrication plant there is a SNRI regime with 24 hours notice and 48 hours retention time for the feed and produced products. There is also a mailbox system with daily information on the production and inventory. The Commission plans to have maximum four interim inspections with 24 hour notice and the IAEA might appear unannounced to the operator during these inspections. A week long PIV is planned as earlier.

For the Studsvik research facility and Clab there is an unannounced inspection regime with at least one inspection each. The inspectors shall be granted access within two hours. PIVs are performed as before.

For the LOF:s, CAM:s and other small installations there is planned one inspection in total for all installations with a frequency of 4 to 6 years

The closed down Barsebäck reactors as well as Ranstad uranium recovery facility will have one PIV/DIV each as before.

Complementary accesses will be performed whenever the IAEA finds it necessary.

A consequence of introduction of UI to Clab and 48 hours SNRI for the reactors was that Sweden proposed to split the site containing both Clab and Oskarshamn reactors into two sites. The reason was that there is a right for the IAEA to ask for a 2 hour CA during an inspection and that would mean that the reactors could be subject to CA when Clab is inspected. This split is now accepted and the update of the AP-declaration this year reflects that situation and there are now 9 sites in Sweden.

Administrative procedures at SSM for integrated safeguards

As soon as the letter from the IAEA confirming the instruction of IS in Sweden was received, the SSM started to launch procedures to be able to participate during the inspections. The receipt procedures of the inspection notifications were changed. A dedicated phone line was selected where all notifications, not only UI, SNRI and CA, is received to a server that distributes the message to two mobile phones, to a dedicated email address and a fax. This phone number was then communicated to the IAEA, the Commission, safeguards staff at Studsvik, Clab and Västerås.

After Sweden joined the European Union 1995, it has not been a legal obligation for the state to participate in the international inspections and this fact became obvious to the Swedish government when IS now was introduced in Sweden and it was clear that the Commission would not be able to participate in all inspections. So the Government decided that Sweden should be represented by SSM at all IAEA inspections.

A rolling scheme has been set up involving all 8 safeguards staff in the non-proliferation section so one inspector has one week at a time starting at noon on Fridays. This inspector has to be prepared to go for inspection immediately when an advance inspection notice is received. One of the staff has the role to coordinate the activities when a notice is received, that is to arrange for a rental car if needed, communicate with the operator and fax or email necessary information for inspector who has already left. Studsvik and Västerås are possible to reach within the two hour limit but for Clab it will take about five hours. So for Clab the agreement with the facility is to let the IAEA inspector in to the facility and, if needed let the IAEA-inspectors be present to watch any process that has to finish and then freeze the situation until the SSM inspector arrives. IAEA can start with paper work as soon as that is available. But since we don't have any experience yet the procedures will have to be adapted to the most convenient way for conducting an UI.

The administrative procedures of SSM was tested almost immediately as a 24 hour CA notice was received already 10 days after IS had begun in Sweden. That was for the Oskarshamn site with the three reactors and Clab.

Conclusions

Coming to integrated safeguards has been a long process and some times difficult as there are so many different parameters to take into account. But it has also bee very interesting and challenging for us who have been involved. The co-operation between the different actors has improved a lot during the process a paved the way for a smooth implementation of IS. For Sweden it also gave us the chance to document the nuclear history, more actors have been introduced to safeguards through involvement in different projects when looking for other tools to be used in safeguards like use of open source information, satellite imagery etc. But we have just started and we still have a long way to go until we have a solid safeguards system in place. There are still states that need to adhere to the additional protocol but we who now have the system in place can be the good examples for those who still are in the decision phase.

Synergy between the French Authority, its technical support body (IRSN) and the operator EDF, to develop authorization files

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Abstract:

For each nuclear facility, the French Authority is issuing authorization for nuclear material holder, based on the organization and the dispositions described into a document: the authorization file. The information contained in this document is representing objectives the operator will follow in order to protect the nuclear material. This authorization file is also the frame of references for controls performed by the Authority. According to French regulations, an authorization file is required both for physical protection and nuclear material control and accountancy areas.

For Nuclear Material Accountancy and Control (NMAC) part, this file contains a description of the facility, it also specifies how nuclear material is followed from its reception to its expedition, as well as the accountancy organization and methods in use to enforce French regulations (Code of Defence).

Among the three main French operators, EDF has the particularity of operating power plants which apply similar NMAC's processes. In 1996, a model of NMAC's authorization file was created by EDF and accepted by the Authority. Based on this model, each site writes its own file. Taking into account site or organization's evolutions and also improvements required as internal control and authority's inspections follow up, files have been frequently updated. Furthermore, improvements requested by the Authority may involve more than one sites in case of generic measures or if the EDF's policy specific aspects are impacted.

The need to update the actual model by creating a new generic authorization file model usable for all EDF nuclear sites was a common understanding for the Authority, its technical support body IRSN as well as for EDF central services. This work will allow to retrieve obsolete chapters and to add new matters of concern. Therefore, the organizational aspects of the plants shall be described in specific "sites files".

A working group, bringing together the Authority, IRSN and EDF has been set up. The first objective was to define an appropriate level of description of common processes in place on sites (reception, expedition, physical inventory, accountancy, follow up of secondary material flow,…). The second objective will be to focus on the level of description of on-site specific arrangement to be implemented. This paper describes the steps followed by this working group, improvements achieved for the different processes description, the difficulties encountered and the work still to be done.

Key words: Authorization file - nuclear material holder - EDF nuclear power plants authorizaton file working group - Processes description

1. Introduction

Electricité De France, owner of 58 reactors established on 19 Nuclear Power Plants (NPPs), is submitted to the French regulation as well as to EURATOM regulation.

For each nuclear plant, the French Authority is issuing authorization for possession of nuclear material, based on the organization and dispositions described into an authorization file. As EDF is operating 19 NPPs, there are 19 authorization files (corresponding to the 80 basic technical characteristics (BTC) required by EURATOM). Therefore, these NPPs possess the particularity of applying similar NMAC's processes.

In 1996, a model of NMAC authorization file was written by EDF and accepted by the Authority. This model was based on the concept of an independent NPP responsible of its own organization and own processes (close to the EURATOM concept of BTC). Each nuclear power plan had then adapted the model to its own specificities.

During more than 10 years, these documents have evolved, but now this concept of independent NPP is too far from the reality of the EDF organization and the need to update the actual model to take into account the global EDF NMAC management system became a common understanding for the Authority, its technical support body IRSN as well as EDF central services.

The purpose of this paper is to:

- Present the French regulation and associated NMAC authorization files;
- Present the EDF NMAC system;
- Describe the experience feedback of the Authority (and IRSN) from control inspections at EDF sites and from analysis of EDF's NMAC documents;
- Highlight the steps of the Authority/EDF/IRSN working group in charge of the revision of NMAC authorization files, the main conclusions and prospects.

2. French regulation and NMAC authorization files

On behalf of the Ministry of Energy, the French national regulations in the field of nuclear materials protection and control aim to provide assurance that proper measures are taken on the French territory to prevent cases of theft, diversion or loss of nuclear materials, and to rapidly detect such cases, should they occur. The base of these regulations is the French Code of Defence (articles L1333). This code is complemented by a set of orders and ministerial instructions.

The Code of Defence makes the procurement of a license obligatory to hold or conduct activities with nuclear materials. It also sets up an administrative and technical control system for nuclear materials, and defines a set of severe penalties to be applied for a number of incriminations including malevolent actions, illicit acquisition of nuclear materials but also failure in warning the police within the 24 hours following the detection of such an event.

The prime responsibility in terms of protection and control of nuclear materials rests on the companies holding the materials. Operators are committed by law to fulfil regulatory objectives and to warn the police immediately whenever they detect a malevolent act or a loss. The objectives of protection and control are set up in several ministerial orders. They are based on an integrated approach combining measures of physical protection, control and accountancy. A specific order dated March 16th, 2004, describes the NMAC measures that have to be implemented.

Requests for nuclear materials licence, submitted by operators to Authority, must be accompanied by an organisational and technical file describing, in particular, the provisions implemented to meet regulatory requirements. This authorization file is required for each nuclear facility to describe both NMAC aspects and physical protection aspects. The information contained in this document presents the dispositions that the operator will follow in order to protect the nuclear material. The descriptions provided by the operator in his authorization file are considered as commitments. The Authority relies on the technical support of IRSN to assess files.

The outcome of the authorization file assessment process is the key base of the Authority decision for granting a license. Specific recommendations can be made to the operator within or at the end of the licensing process.

The control of the implementation of this regulation by the Authority is made through regulatory inspections conducted in the licensed facilities by sworn and accredited agents. The content of this authorization file in addition to the content of regulations are the frame of references for inspections performed by the Authority.

The assessment process and the inspection process work closely together and are complementary. The combination of assessment and inspections allows the controllers (Authority and its technical support body) to apprehend technical particularities and accompany operators, when needed, in an improvement cycle requiring their direct involvement.

In order to fulfill the regulations, the licensee has the legal obligation to know at any time the quantities and qualities, the locations, uses, movements and transformations of nuclear materials in his facility. The NMAC part of the authorization file has to contain, first a general description of the plant, then a description of:

- The plant's follow up prescriptions from the former reception of nuclear material to its expedition;
- Physical inventory prescriptions in use;
- Accountancy methods in use.

3. Nuclear Material Accountancy and Control management system at EDF

EDF operates 58 PWR reactors established on 19 sites. To take advantage of these standardized units most of functional tasks and some of operational tasks are assumed by central common units.

So the existing NMAC system at EDF was built on a compromise between on one hand standardization and uniformity, headed by functional departments at central level, and on the other hand of independence and responsibility of head managers of nuclear power plants.

The main Divisions Concerned by the NMAC system inside the "Direction Production Ingénierie" (Engineering and Operations Division are:

- Division Production Nucléaire (Nuclear Operations Department) composed of:
  - 19 Nuclear Power Plants on the French territory, responsible of Nuclear Material Accountancy and of the follow up of each object which contains Nuclear Material;
  - one unit located in "Lyon", responsible of Nuclear Transformations Calculation, and core loading maps;
  - one unit located in "Paris" responsible to determinate the Nuclear Material protection rules to apply in each plant;
- Division Combustible Nucléaire (Nuclear Fuel Department) located in "Paris", composed of:
  - a department in charge of the nuclear fuel supply for the NPPs;
  - a department in charge of organizing the removal of irradiated fuel for the NPPs and its transportation to the AREVA reprocessing plant of La Hague;
  - a department responsible

- to determine the NMAC rules to apply in each plant to satisfy French and EURATOM regulation and to draft the generic parts of the authorization files for each NPP,
- to assist each unit to implement safeguards,
- to define and control the functional rules implemented in the EDF Computerized Nuclear Material Accountancy System,
- to assist the Nuclear Operation Division to define the training for NMAC field,
- to be the EDF technical contact for the authorities and their supports;

- Division Ingenierie Nucléaire (Nuclear Engineering Department) made up of:
  - one unit under decommissioning (“Creys-Malville”);
  - one unit under construction : EPR (“Flamanville”);

- Division Appui Industriel à la Production (Production Support Department), composed of:
  - one unit located in “Bordeaux” responsible to develop and maintain in operational conditions the EDF Computerized Nuclear Material Accountancy System;
  - one unit located in “Châlon-sur-Saone” in charge of the training in the field of NMAC;
  - one unit based in "Pacy-sur-Eure" in charge of operating the EDF Computerized Nuclear Material accountancy System.

In order to ensure uniform interpretation of regulations and practices, the Nuclear Fuel Division is in charge of establishing adequate guidelines and assisting the local operating units in implementing safeguards.

The actual NMAC documentation system is based on the generic guidelines draft by Nuclear Fuel Division, approved by Nuclear Operations Division, but not by authorities. These guidelines are used by each NPP to draft its own operational documents and its own authorization file, each file is send to Nuclear Fuel Division to be checked before sending to the French authorities. The French authorities may ask the plant to modify the file. Theses modifications must be integrated in the authorization file of the NPP and, if needed, in the generic guidelines, and then in each authorization file.

This system needs to be optimized and needs a better integration in the operational organization. The first step of this evolution is to develop guidelines not just to answer to the French requirements but also to be as close as possible to the operational processes applied in the NPPs. The objective is to demonstrate to the authorities that the industrial process applied is a correct answer to NMAC requirements. This job began in 2006.

4. Experience feedback from the Authority (HFDS) and its technical support (IRSN)

4.1 Feedback from French national inspections (and EDF’s answers to Authority requests)

As previously mentioned, the aim of inspections is not only to verify that regulations are respected but also that the commitments made by the operator in its authorization file comply with the reality. Inspections cover technical and organizational provisions and consist in an on-site examination of the measures taken by EDF and determination of compliance with general or specific requirements. They represent a second level of control since operators remain the mainstay of the protection and control exerted over nuclear materials. Considering the fact reported by inspections, the Authority send requests to the operator.

For the last five years, most of the requests of the Authority concerns physical inventory taken dispositions, transfer of information between the follow up level and the accountability level, but also, for instance first level of control in unforeseen events as well as re-expedition of fresh fuel in case of non-conformity.

Furthermore, most of the improvements required by the Authority after an inspection don't necessarily involve a particular nuclear power plant but may be generalized for all the sites. In fact, most of the time, the "policy" aspects of EDF is impacted.

4.2 Assessment of updated authorization files

Since 1996, each one of the 19 EDF's authorization files has been updated several times, mostly in order to take into account:

- Frequent changes in the organization of the plant;
- Improvements implemented by the plant to take into account demand from the Authority issued after inspections;
- Feedback based improvement cycle.

After each modification, the authorization file is sent to the Authority and assessed by its technical support IRSN. The aim of this assessment which cover technical and organizational provisions, is to verify that the measures taken by the operator comply with regulatory documents and that the requests of the Authority are taken into account. Inspections complement this assessment by verifying that measures to be taken are effective on the field.

However, most of the improvements requested by the Authority involve more than one site and can be turn to generic improvements.

4.3 Assessment of annual physical inventory reports

The French regulation indicates that a licensee has to set up a physical inventory of all the nuclear material it holds, at least once a year. Furthermore, a "physical inventory report" has to be written by the operator and sent to IRSN within 45 days.

The content of a physical inventory report has been detailed in a guideline sent by the Authority to every nuclear material licensee in 1995. It contents an abstract of the implemented procedure for physical inventory taken, a comparison of the data recorded by the French national centralized accounting database and the operator's local accounting data, as well as a comparison of the results of physical inventory taking and the accounting data. Discrepancies and resulting corrective actions have to be indicated in the report.

The particularity of EDF is that nuclear materials are confined in assemblies, with apparently no possible inventory discrepancies. Nevertheless, some disparities could occur, for example lack in identification, error in the localization of assemblies or however the hypothesis of a missing assembly must be foreseen.

Most of the 58 annual reactor physical inventory reports require a precise description of the inventory taking conditions as well as a description of the difficulties encountered. This generic aspect should be included in a same physical inventory procedure for all EDF NPPs.

4.4 General feedback and best practices

The feedback from analysis ran by the authority and its technical support body of authorization files, as well as the fact reported from inspections have to be investigated from a general point of view, including all the facilities and not only EDF's nuclear power plants. In fact, in order to implement the best practices, new or more complete aspects should be considered by the operator, in particular for the following item:

- Security of information management system, with an appropriate description of the prescriptions implemented in order to avoid any data falsification or destruction;

- Crisis organization, with a description of the organization and resources to be implemented in order to manage a quick inventory, in case of crisis involving a suspicion of nuclear material loss;

- Internal verification and quality audits, with precisions about the type and frequency of such controls.

5. Implementation of an Authority, EDF and IRSN working group

5.1 Objective

The necessity to create a generic authorization file for all the EDF NPPs and to include new aspects inside became then particularly clear.

However, sites organization and specificities still have to be described in specific “sites files”.

In order to perform this task, a working group bringing together the Authority, IRSN and EDF has been set up.

5.2 Implementation of the working group

The working group was created on February 2008, by a letter from the Authority, sent to EDF and IRSN.

During a first meeting which took place two months later, IRSN presented the list of the main improvements to be considered.

Then, EDF listed the documents, about ten, which would constitute the generic authorization file. Each document describes a process: from the reception of nuclear material to its expedition, including physical inventory taking as well as accountancy dispositions.

The planning of the working group was then established with an objectif at the end of March 2009 to produce a generic authorization file.

5.3 Working group method and planning

The working group met once a month and analyzed one process by month. Before, every meeting, one EDF’s member of the group submitted a draft of a document and the members of IRSN listed remarks and improvements to be considered. Then, the meeting devoted itself to discussions on improvement suggestions. Finally, a new version based on the conclusion, is being draft by EDF.

More precisely, the processes analyzed by the working group cover three aspects:

- Main principles and characteristics, covering “general principles”, “official documents” and “characteristics of the nuclear material’s items involved”;

- Main physical operation on fuel assemblies, i.e. “reception”, “expedition”, “re-fuelling”, “Fuel assemblies restoration” as well as “physical inventory taking”;

- Main provisions on accountancy: “accountancy tasks required at each end of the month”, “error corrections and unavailability of the computerized application”.

The first step was reached at the beginning of April 2009, after a two days meeting of the group where all the processes have been examined.

5.4 Example of a particular document: General principles

This document is the main one which organized and makes the links between the others.
The objective of “general principles” document is to present the principles adopted by EDF in order to fulfill regulation. This document introduces also the set of documents which constitutes the generic EDF’s authorization file.

More precisely, this document specifies, first, the prescriptions required by EDF in order to set up an authorization file. It is important to notice that it includes also the EURATOM requirements.

The second part of this document indicates the relation with the French’s Authority and its technical support IRSN. After a description of the main tasks assigned to the EDF’s Departments (see §3) organization, the EDF’s liabilities are described: the liabilities of EDF’s “Specially appointed Representative” for NMAC point of view, which is responsible of the nuclear materials present on all EDF’s NPPs, then EDF’s “Plant appointed Representative” which is the Director of the plant, and finally, the “Nuclear Material Holder Representatives” chosen between the executive team of the plant. One “Nuclear Material Holder Representative” must be available at any time. He must be able to take any decision necessary in case of suspicion of loss or discovery of nuclear material and must warn the Authority “as soon as possible”. Finally, staffs in relationship with the French national accountancy, kept by IRSN, are identified.

In its third part, it gives fundamental principles about the follow up and more precisely the follow up methods used for assemblies, fuel rods and other items that contain nuclear material. The prescriptions for accountability are then detailed, with a description of bookkeeping related actions as well as declarations due to the French national accountancy. Finally, the information management system, its functionalities and protection are described.

The fourth part of this document described the professional training required for staffs involved in NMAC.

The fifth part deals with knowledge-sharing at EDF.

The following two parts are devoted to quality and internal verifications (including audits) and to document storage rules (for NMAC records).

The last part deals with the content and the management of this set of documents that forms the generic EDF authorization file.

The working group identified two items to develop in the second version: the EDF’s objective in term of NMAC and an introduction to the EDF’s security culture.

5.5 Examples of significant improvements

During working sessions, the group raised some significant improvements to be implemented in the set of documents; some of them are listed below:

- The way to inform the Authority in case of significant difficulty related to the follow up, the operator will send a fax to the Authority (a model is attached to the concerned documents);

- The process to follow in case of unavailability of computerized application, for each main case of unavailability (one site unavailable or all sites). The operator has to send a fax to the French national accountancy (a model is attached to the concerned documents). The description of controls to be held at the end of the unavailability;

- The identification of the main dysfunctions that could appear for each process. The benefit of this identification is to keep in mind the risks, in order to prevent or correct more efficiently any eventual dysfunction ;

- The importance of internal verifications is raised for each process. The objective of these controls was also highlighted in order to remind the risks and the necessity of improvement;

- For each type of document (NMAC technical justification document), the archiving time was clearly established;

- The different stages of the control were defined when assemblies receipt: the first level of control (required by French regulation within one working day) and the final one which allow the receiver to check the conformity of the objects;

- The traceability requirements of the follow up operations were clearly established in the specific case of fuel rods extractions and/or insertion.

5.6 Difficulties encountered

Even if the working group didn't have to face any particular problem, there were few difficulties to overcome.

Actually, it was not so easy to determine prescriptions suitable for the 19 licensees and acceptable by the Authority. As the French regulation is based on a performance based approach, the Authority has to know and understand EDF's NMAC objectives implementing the regulation whereas the sites need to know unambiguous prescriptions to follow.

From the Authority and its technical support body's point of view, the accountability system was more detailed than the follow up system. A more balanced description was needed.

Finally, from EDF point of view, the difficulty was to find an agreement between the industrial constraints and the NMAC concepts of French and EURATOM authorities.

5.7 Job to be done

After this first step, the writing of a guideline for sites authorization files remains to be done.

In order to write this guideline, a second working group will be set up, with representative actors from the sites.

At the end of this second step both generic authorization file and sites specific files for each of the 19 sites, will be sent for approval to the Authority.

The final objective is to get all the authorization files approved before March 2010.

A meeting to analyze the feedback of this working method will be held by the end of 2010.

6. Conclusion

One year after the creation of the Authority/EDF/IRSN working group, the assessment of the duty performed by the group is clearly positive.

The whole set of documents forming the generic authorization file have been examined by the group. This set of documents is describing the Nuclear Material Accountancy and Control common processes implemented at EDF's Departments and on Nuclear Power Plants, taking into account best practices, experience feedback from both Authority and EDF internal control. The second part of the work shall allow to reach the final objective before March 2010. So, it would have taken respectively four years

for EDF and two years for the working group (from 2006 to 2008 for EDF internal work, from 2008 to 2010 for validation with the Authority), to review all EDF NPPs authorization files

The benefit of this method is multiple. It allowed:

- A common better understanding between the Authority (and its technical support) and the operator, leading to a proper integration of the view of both of them;
- The implementation of the global approach of EDF's NMAC organization resulting of a "top to the bottom" methodology;
- A better identification of the responsibilities between EDF's departments (head office) and nuclear power plants.

Finally, after the completion of the first step, the achievement of the working group's tasks, is definitely linked to the motivation of each working group member to implement the best NMAC practices. All together, the team leader from the Authority, the four members from EDF, and the two members from IRSN show a strong involvement working jointly to achieve this goal.

The next step for EDF will be to try to have a similar approach with EURATOM, taking advantage of this work.

Evolution of Safeguards Analytical Services

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Abstract:

Safeguards analytical services began in the 1970s with the objective of verifying State declarations regarding nuclear material accountancy via independent analysis of safeguards nuclear material samples collected by the IAEA. Analysis focused on uranium and plutonium concentrations and isotopic abundances. Analytical results were evaluated statistically to assess the correctness of declarations regarding nuclear material inventory and flow. In the 1990s, during the process of strengthening and streamlining IAEA safeguards, the need for analytical techniques to detect indications of possible undeclared nuclear activities or undeclared nuclear material was identified. Environmental sampling techniques were implemented and further developed by the IAEA with support from Member States. Today swipe samples are regularly collected during inspections, design information verification, and complementary access. The analysis of swipe samples is performed on the particle and bulk level. The presence of trace levels of uranium and/or plutonium may indicate past or existing nuclear operations, and their isotopic composition may indicate specific nuclear fuel cycle activities. Other analytical techniques have been used when required but have not yet been implemented for routine application. Such future techniques may include highly sensitive spectroscopic analysis of radioactive nuclei produced by nuclear operations and their decay products, impurity analysis of nuclear material samples, and analysis of specific elements which may be characteristic of certain nuclear activities. This presentation reviews the analytical techniques used in the past and today and identifies those which may be used in the future. Currently the IAEA has initiated a project for upgrading and modernising its Safeguards Analytical Laboratory in Seibersdorf, Austria. The outline of the project and the conceptual design of the new laboratory are also presented.

Keywords: safeguards analysis, environmental sampling, additional protocol, safeguards analytical laboratory.

1. Introduction

The challenges facing the IAEA concerning nuclear material safeguards (SG) have changed over time and the IAEA Safeguards System has continuously evolved to meet these challenges. For example, in response to these challenges, the types of inspection activities applied, the types of samples taken, the laboratories and analytical techniques employed, and the types of evaluations performed have all evolved during the past several decades. This paper will focus on the evolution of SG analytical services, which include destructive analysis (DA) techniques involving bulk nuclear material (NM) and environmental sampling (ES). In particular, an increasingly investigative approach is applied by the IAEA to maximize the amount of information obtained from sampling activities, which may reveal the presence of possible undeclared nuclear materials or activities.

1.1. History

Destructive analysis of NM began in the 1970s, during which Member States' laboratories analyzed NM samples taken by the IAEA. In 1975, the IAEA Safeguards Analytical Laboratory (SAL) was established in Seibersdorf, Austria, at which point the IAEA began performing its own analytical measurements of NM samples to verify State declarations. During the 1980s, SAL's capabilities, as well as those of other laboratories in the IAEA's Network of Analytical Laboratories (NWAL) expanded. Environmental sampling was introduced in the 1990s after the discovery of a clandestine nuclear weapons program in Iraq indicated a need for new measures to detect undeclared activities. The usefulness of ES was confirmed during a series of field trials in the development programme to

strengthen safeguards, entitled “Programme 93+2”. In 1996 the SAL Clean Laboratory (SAL/CL) was established. A timeline of some major milestones in the history related to IAEA SG analytical services and some key inspection activities are provided in Table 1.

Date	Milestone(s)
1970s	Destructive analysis of NM begins (Member State laboratories perform analysis)
1975	SAL established in Seibersdorf
1980s	SAL capabilities and NWAL for NM expand
Early 1990s	IAEA obtains initial experience using ES in Iraq Strengthening safeguards (Programme 93+2): ES field trials show usefulness of ES
1995	IAEA Board of Governors (BOG) approves ES as a new SG measure under existing legal authority of SG agreements
1996	IAEA SAL Clean Laboratory (SAL/CL) established ES begins as activity under SG agreements
1997	IAEA BOG approves Model Additional Protocol providing broader use of ES during complementary access
1998-present	NM samples and ES activities continue Samples taken in a large number of States including special sampling campaigns in Iraq, Libya, Iran, and Syria

Table 1. Some Major Milestones in the History of Analytical Services

1.2. Broadening of SG Verification Measures

A major milestone in the evolution of SG analytical services was the broadening of SG measures to include not only the verification of declared nuclear materials, but also the detection of undeclared nuclear materials and activities. While previously (in the 1970s and 1980s) sampling primarily focused on DA samples taken for verification of declared NM inventories and flow (for material balance purposes), during the 1990s the IAEA implemented new sampling and analytical techniques which are useful for detecting undeclared nuclear materials and activities.

1.3. Laboratory Infrastructure

Destructive analysis of NM samples taken by IAEA inspectors began in Member State laboratories in the 1970s and in particular at SAL in 1975. Since then the IAEA has expanded and adapted its laboratory infrastructure, including new sample processing areas and analytical instruments to accommodate the types of analysis and numbers of bulk NM samples collected. In 1996, the SAL/CL was established, which has become the focal point for the management of ES samples (including screening, initial processing, and archiving) and which maintains a variety of primary analytical techniques employed in the ES area. In addition, various laboratories around the world have joined the IAEA’s NWAL for both NM and ES analysis, which has the goal of providing high quality analytical services, diverse analytical techniques, and increased sample analysis capacity (both for routine sample collections and “surge” situations during large and often high priority sampling campaigns). In particular, the expansion of the NWAL has made it possible to analyze replicate environmental samples by two or more laboratories, which employ various processing and analytical techniques. While the addition of the ES program has been the most dynamic change, the tasks associated with processing and analyzing bulk DA samples have also grown in complexity, and the need for new analytical equipment and additional laboratory space related to bulk DA sample analysis has also grown. The current IAEA project titled “Enhancing Capabilities of the Safeguards Analytical Services” (ECAS), which has the goal of addressing new infrastructure and analytical challenges related to SG analytical services for both bulk NM and ES samples, is also discussed in this paper.

1.4. Areas of Growth – Application of New Analytical Techniques and Approaches

As the IAEA’s analytical needs have grown, SG analytical services have adapted to provide the necessary analytical support or facilities. Some of these areas of support include:

- Development of capabilities to get additional information from U DA samples taken (besides U concentration and enrichment) via trace-level impurity analysis and isotopic measurements for

the minor isotopes of U and isotopic analysis of elements such as Pb, O, etc. These U containing materials include uranium ore, yellowcake, uranium fluoride (UF), and other U compounds. Trace-level impurity analysis may reveal the origin of source material and/or processing history;

- The implementation of advanced particle analysis techniques, such as Ultra-High Sensitivity Secondary Ion Mass Spectrometry (UHS-SIMS) for ES samples. This technique should solve several key measurement problems inherent in conventional SIMS instruments and provide U minor isotope performance which is comparable to the Fission Track Thermal Ionization Mass Spectrometry (FT-TIMS) technique but at the same time maintain the advantages of conventional SIMS (such as simplified sample processing, analytical flexibility, and elemental analysis capabilities);
- Analysis requests requiring special nuclear forensic analysis – these requests may involve elemental, chemical, and/or isotopic analysis.

These techniques require increasing investment in sampling, instrumentation, evaluation tools, and/or research and development activities.

2. Broadening of SG Verification Measures

Before various challenging inspection situations in States such as Iraq and the Democratic People's Republic of Korea (DPRK) in the early 1990s, the IAEA focused primarily on verification of declared inventories of nuclear materials and related activities. Some of the key verification methods used since the beginning of IAEA SG include:

- DA sampling and subsequent measurement at SAL and/or Member State laboratories;
- On-site NDA measurements;
- Item verification (presence and ID check);
- Containment and surveillance (such as application of seals and installation of surveillance cameras);
- Review of nuclear accountancy records (including inventories, flow, etc.);
- Material Balance Evaluation (primarily focused on detection of possible diversion of U or Pu).

These verification methods have proven to be effective for verification of declared nuclear material inventories and flow, but additional verification was needed to detect undeclared activities in situations such as:

- Past activities at locations where access is available but where bulk materials and/or processing equipment have been cleaned or removed;
- Current or past undeclared activities in adjacent and/or inaccessible rooms or buildings - in this case samples taken in adjacent rooms or buildings, or in the environment outside those buildings, may reveal trace quantities of material which have escaped from processing or storage areas;
- Nuclear-related processing equipment of unknown origin - for example, equipment declared as "new" and/or "domestically produced", may contain trace levels of nuclear signatures and indicate undeclared sources of nuclear-related equipment;
- Undeclared sources of NU materials for conversion activities - in addition to material balance efforts, trace level impurity analysis may provide an additional way to detect undeclared U materials (in the case of NU, U isotopics normally cannot be used to differentiate sources or batches).

In order to develop capabilities to verify that undeclared activities have not occurred, the IAEA first investigated ES techniques (Programme 93+2), by taking various types of samples (soil, water, vegetation, swipes) within and outside declared nuclear facilities in various Member States around the world. The conclusions drawn included the following:

- Signatures found on samples taken at or near facilities generally matched or revealed the types of operations occurring at the facilities;

- Signatures in the environment in many cases could be seen at distances many kilometers away from the facilities;
- On-site swipe sampling combined with particle analysis was the most effective sampling and analysis technique.

Based on the results of the Programme 93+2 field trials, the IAEA's BOG approved ES as a new SG measure in 1995, and the first environmental samples were taken in 1996. Under the Agency's existing legal authority in SG agreements, swipe samples were collected in facilities and at locations where the Agency had access during inspections and design information verification (DIV) visits under existing arrangements. The approval of the Model Additional Protocol by the BOG in May 1997 and its subsequent adoption by safeguarded States broadened the scope and legal authority of Safeguards by providing the Agency with greater access to nuclear fuel cycle related information and locations. Under an additional protocol, the collection of environmental samples is explicitly defined as an activity that can be conducted during complementary access (CA) at a broad range of locations.

Samples taken during inspections and DIV visits are most often from established nuclear facilities such as enrichment plants and facilities with hot cells. Samples taken under CA are taken at a more diverse set of installations, including established nuclear facilities, but also at universities, research centers, and many locations which can have a broad range of operations and may have little or no inventory of nuclear materials. The implementation of ES under CA has resulted in more diverse samples which often require a larger set of sample processing methods, analytical techniques, and evaluation tools, creating significant challenges for SG. A summary of some of the primary SG verification methods implemented since 1970 is shown in Table 2.

Approximate Time Period	Cont. and Surv.	On-Site NDA Meas.	DA Sampling	Material Balance Evaluation	ES Sampling	Impurity Analysis of Bulk NM
1970-1975	X	X	X	X		
1975-1980	X	X	X	X		
1980-1985	X	X	X	X		
1985-1990	X	X	X	X		
1990-1995	X	X	X	X	X*	X*
1995-2000	X	X	X	X	X	X
2000-2005	X	X	X	X	X	X
2005-pres.	X	X	X	X	X	X

*Various new sampling and analysis methods were first tested during inspections in Iraq and/or the Programme 93+2 field trials.

Table 2: Broadening of Safeguards Verification Methods

2.1. An Investigative Approach

A key feature of the evolution of SG analytical services is the increasingly investigative approach which is applied. Investigative analysis of ES swipes, other environmental samples (such as air filters, soil, vegetation, etc.), and NM samples (such as U compounds) allow the IAEA to look for particular clues or indications of small amounts of material which may indicate possible undeclared activities, past or present, even in cases where materials and/or facilities are hidden or equipment is cleaned.

Concerning Investigative Approaches involving ES Techniques (Swipe Samples):

- U minor isotopes are important in understanding the type(s) of feed material and the type(s) of enrichment processes which may have been used, or in detection and characterization of irradiated materials and irradiation scenarios, etc.
- Pu isotope abundances are important, not only for assessment of burnup and possibly verification of the use of a specific type of reactor, but also for determining reactor and chemical processing history, using trans-uranium isotopes such as ^{241}Pu and ^{241}Am for age-dating purposes.

- Various gamma-emitting radioisotopes also help identify the specific activities which have occurred, giving information such as types of materials irradiated, time since irradiation, and whether or not chemical separation has been performed, etc.
- The elemental and/or chemical composition of various materials at a facility may reveal the presence of key nuclear and/or non-nuclear materials and help determine consistency of findings with declarations.

Concerning Approaches Involving Other Types of ES Samples (soil, vegetation, air filters, etc.)

- Sensitive $^{236}U/^{238}U$, $^{129}I/^{127}I$ measurements, and precise $^{235}U/^{238}U$ measurements may reveal perturbations from natural or established background concentrations in the environment, which may provide indications of clandestine activities involving enrichment, reprocessing, reactor operations, etc.
- Air filter samples or other media may contain interesting (enriched, radioactive, etc.) particles at a distance of kilometres from facilities and also cover relatively large areas, where ES swipe sampling may not be realistic due to the large number of buildings and facilities.

Concerning Investigative Measurements of NM Samples

- Measurement of trace-level elemental impurities, O and Pb isotope ratios, as well as other analytes may provide key SG information about sources or processing history of materials.
- Since much of the nuclear fuel cycle involves U compounds with natural isotopics (i.e. isotopics found in environmental U), these types of additional information may provide the best methods for confirming consistency with declared source(s) and processing of NU compounds.

These approaches require state of the art sample processing techniques, clean laboratory facilities, analytical instruments, staff training, sample management, and NWAL management. These tasks may also be challenging due to the need to maintain high sample capacity levels, to satisfy IAEA inspection requirements.

2.2. Analytical Support for ES

After ES sampling methods were approved and construction of the SAL Clean Laboratory was completed, the IAEA began taking ES swipe samples on a regular basis and other types of environmental samples from time to time. This has required the development and implementation of support functions which include preparation of sampling kits, inspector training, the application of appropriate analytical techniques at the NWAL, and expansion of the NWAL.

2.2.1. Sampling Equipment and Support

Sampling kits have been developed to satisfy three major sampling requirements:

- Sampling at enrichment facilities or locations outside of hot cells (where high activity levels of radioisotopes are not normally expected);
- Sampling inside hot cells where high activity levels of radioisotopes are expected;
- Pre-inspection check samples to assess the level of any U or Pu contamination present on inspectors' hands and clothing before the inspection (such a sample is taken before entering the facility).

Inside each sampling kit are a given number of replicate swipes and equipment for sample taking, as well as a copy of the sampling procedure. Over time, the sampling kits and procedures have been improved.

A summary of the approximate number of ES samples taken since 1996 is shown in Figure 1. The average sample load per year is seen to be about 400, with a significant increase occurring in 2003-2005 due to various large sampling campaigns.

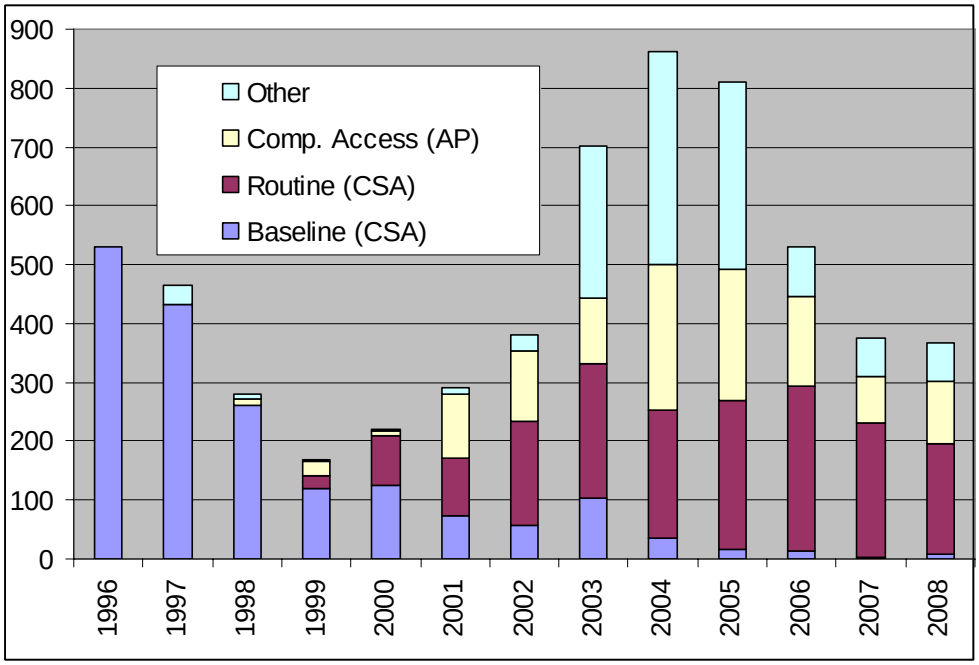

Figure 1: ES Samples taken from 1996-2008. A significant portion of ES is now taken under CA.

2.2.2. Analysis Techniques

Techniques used to analyze ES (typically swipes) can be generally classified into particle and bulk techniques. Since the mid 1990s, the following primary techniques have been used (see Table 3):

Technique	Part.	Bulk	Typical Applications
Optical Microscopy	X		Isolation and preliminary characterization of particles
Fission Track Thermal Ionization Mass Spectrometry (FT-TIMS)	X		Isotopics of individual particles, qualitative analysis of Th content and the U/Pu ratio
Secondary Ion Mass Spectrometry (SIMS)	X		U Isotopics of individual particles (operated in search mode for particle location and preliminary identification, followed by microprobe analysis for more accurate isotopic analysis). Elemental analysis can also be performed on individual particles across a wide mass range.
Scanning Electron Microscopy (SEM)	X		Elemental and chemical composition; various detectors and additional equipment to perform specific types of analysis can be applied
Isotope Dilution Mass Spectrometry (IDMS)		X	U and Pu concentration and isotopics
Thermal Ionization Mass Spectrometry (TIMS)		X	U and Pu isotopics
Inductively Coupled Plasma Mass Spectrometry (ICP-MS)		X	U and Pu isotopics; Bulk analysis of environmental samples; Bulk NM samples (such as U compounds) which are taken for investigative purposes can be analyzed for elemental impurity content
X-ray Fluorescence		X	Elemental composition, often used for swipes and bulk samples to characterize the content of U and other elements
High Resolution Gamma Spectrometry (HRGS)		X	Radioisotope activity levels, including radioisotopes produced by neutron activation and fission, as well as various decay products
Accelerator Mass		X	Primarily used for $^{236}U/^{238}U$ and $^{129}I/^{127}I$ Ratios; Pu content and

Spectrometry (AMS)			$^{240}Pu/^{239}Pu$ measurements can also be performed.

Table 3: Common Analytical Techniques used for ES and other investigative Purposes

As the ES program has progressed, increasing emphasis has been placed on high-quality U minor isotope measurements (in particular ^{234}U and ^{236}U) in particles as well as in bulk samples. In addition, efforts have been made to lower detection limits for Pu, particularly in swipe samples to levels of about 10-100 fg. A variety of techniques have been employed for the analysis of environmental samples other than swipes; the most appropriate techniques appear to be bulk MS techniques (U and Pu concentrations and isotopics), HRGS (for radioisotopes activity levels), and AMS (for detection of perturbed $^{129}I/^{127}I$ and $^{236}U/^{238}U$ ratios)

2.2.4. ES Quality Control (QC) Program

A robust quality control program has been implemented to help monitor for potential problems related to the analysis of ES samples. This QC program includes the following:

- Regular analysis of ES samples by at least two different laboratories and two different techniques including HRGS, FT-TIMS, SIMS, and/or bulk analysis;
- Regular distribution of blind QC blank and control samples to NWAL laboratories;
- Regular feedback to NWAL laboratories regarding performance on blind QC blank and control samples;
- Regular distribution of non-blind QC blank samples to NWAL laboratories performing bulk analysis, to help check for process contamination and help estimate laboratory process detection limits;
- Distribution of non-blind QC control samples to NWAL laboratories for performance improvement and development purposes;
- Analysis of selected pre-inspection check samples to look for contamination on inspectors' hands and clothing before entering facilities;
- Periodic determination of detection limits for the various techniques used at bulk analysis laboratories;
- Regular technical meetings where Agency staff can meet with NWAL scientists to discuss analytical performance, QA/QC programs, R&D, and other issues.

2.3. Analytical Support for Verification of Declared NM Inventories and Flow

Support for NM inspection activities has developed over time and has included improvement of sampling equipment and procedures, shipping containers and procedures and, since the inauguration of SAL in 1975, implementation of a host of analytical techniques which allows SAL to perform almost all NM analyses. The IAEA also has implemented advanced NDA equipment and techniques to make measurements in the field, including various instruments used for measurement of item weights, U total, Pu total, and selected U and Pu isotopes.

2.3.1. Sampling Equipment and Support

Four primary bulk NM sample types are taken, namely various U and Pu compounds, heavy water (D_2O), and input solutions from reprocessing facilities (containing U, Pu, and fission products). Although the types of samples taken have not significantly changed, over time the IAEA has expanded the types of sampling equipment (such as sampling bottles of various sizes and made of various materials). In addition, procedures, inspection working papers, and inspector training have been updated to help improve sample taking and provide, as far as possible, representative samples which can be shipped, analyzed, and evaluated as efficiently as possible. A summary of the approximate number and types of bulk NM samples taken since 1996 is shown in Figure 2.

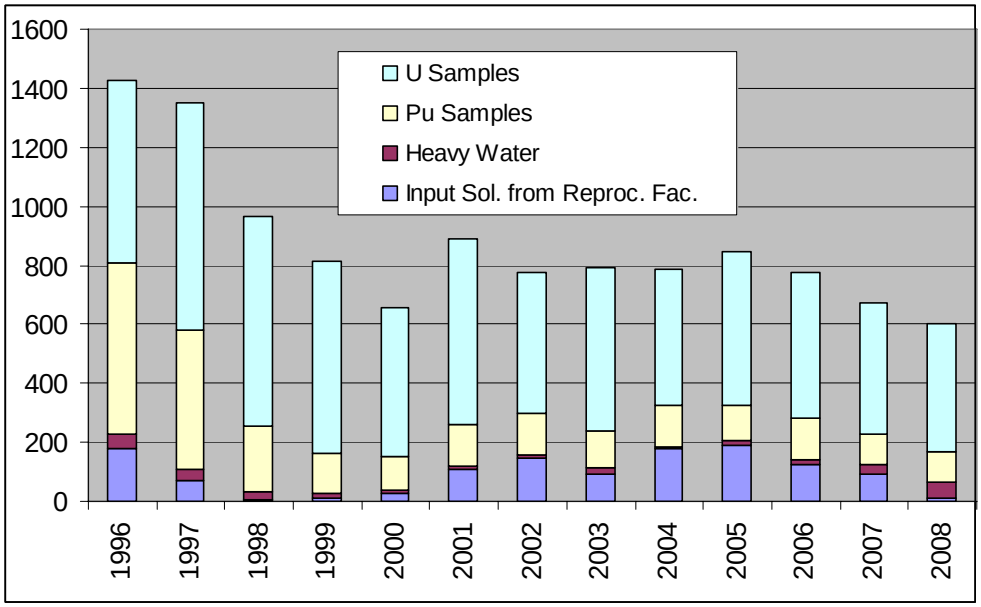

Figure 2: Estimated number and type of bulk NM samples taken since 1996.

2.3.2. Analysis Techniques

Primary DA techniques used in the past by SAL and other network laboratories for U, Pu, and Heavy Water analysis (for Material Balance Purposes) include the following (see Table 4):

Technique	U Conc.	Pu Conc.	U Isotopics	Pu Isotopics	Impurities	D_2O Abundance (Heavy Water)
Titration	X	X				
Gravimetry	X	X				
Coulometry		X				
Isotope Dilution Mass Spectrometry (IDMS)	X	X	X	X		
Thermal Mass Spectrometry (TIMS)			X	X		
Inductively-Coupled Plasma Mass Spectrometry (ICP-MS)					X	
Density Meter; IR Spectrometer						X*

*Heavy water analysis is performed at the Central Research Institute for Physics in Hungary.

Table 4: Primary DA Techniques

Techniques such as ICP-MS are currently being implemented at SAL/CL and have been implemented at many of the other network laboratories. ICP-MS is expected to play an increasingly important analytical role in the future.

3. Evolution of the Analytical Laboratory Infrastructure

As SAL's facilities grow older and new SG challenges arise, the need to plan for future facilities has increased. In response, planning the design and possible future construction of new facilities, and/or upgrading of SAL's current facilities has begun, and is embodied in the ECAS project.

Feasibility studies have been performed including:

- The SAL Workshop (Nov 2006), which brought together expert staff and consultants from Member States to discuss the necessary functions of SAL;
- The SAL Study Group (Mar 2007-Nov 2007), which involved detailed work by the Secretariat and input from consultants who studied the functions of SAL in more detail.

A November 2007 report by the DG to the IAEA BOG set out several goals related to the vision of "Sustaining Credible Safeguards" and enhancing the IAEA's independent analytical capabilities, including plans to:

- Introduce high-sensitive environmental sample analysis: procure and install a UHS-SIMS (Ultra High-Sensitivity Secondary Ion Mass Spectrometry) instrument and extend the Clean Laboratory (Phase 1, high priority, urgent);
- Upgrade and modernize the Nuclear Laboratory: construct and commission a new laboratory (Phase 2, high priority, requiring significant funding and careful planning);
- Enhance the use of NWAL laboratories; and
- Apply all relevant security and safety requirements.

3.1. Evolution of SAL – Enhancing Capabilities of the SG Analytical Services (ECAS)

In response to the challenges facing SAL and the guidance from the SAL Workshop, SAL Study Group, and DG's report, the ECAS project was set into motion.

- In June 2008, the interdepartmental ECAS project team was established;
- Project planning was initiated in July 2008, and included the adoption of international project and quality management standards;
- A circular letter was distributed to Member States with Support Programmes requesting financial and expert support;
- In October 2008, procurement was initiated for conceptual planning and design;
- Offers of voluntary contributions including expert consultants have been received;
- Qualification for NWAL is on-going; and
- Conceptual planning and design has been initiated.

A detailed description of the current ECAS project timeline is as follows:

- Phase 0: Conceptual planning and design of phase I and II
    - Task 1 (2008-2009): Future overall conceptual plan for SAL;
    - Task 2 (2008-2009): Conceptual design of CLE (Clean Lab Extension);
    - Task 3 (2009-2010): Layout and conceptual design of NL (Nuclear Laboratory).

- Phase 1: (2009-2010) Acquisition and installation of UHS-SIMS and SIMS and construction of CLE
    - Task 4: Purchase and installation of UHS-SIMS;
    - Task 5: Engineering design, construction, and commissioning of CLE and infrastructure preparation of CLE.

- Phase 2: (2011-14): Construction and commissioning of the Nuclear Laboratory
    - Task 6: Engineering design, construction and commissioning of NML and infrastructure preparation of NML

3.2. Evolution of the NWAL

The NWAL includes laboratories supporting both the NM and ES programs. Since the vast majority of NM samples are analyzed by SAL, it has not been necessary to expand or adapt the NWAL

laboratories performing NM analysis as much as that of NWAL laboratories performing ES analysis, which has required a more dynamic process.

3.2.1. Early NWAL Support for ES

Early NWAL support for ES consisted of several laboratories (including SAL/CL) performing several types of analytical techniques. SAL/CL's function at the start of the ES program was to receive, screen (primarily by HRGS), to analyze a portion of the samples (primarily by bulk or SEM particle techniques), to process samples for re-distribution to other laboratories, and to archive samples. The main techniques at other laboratories (primarily in the US but also in several other Member States such as France, Russia, and the UK) included both particle (usually FT-TIMS) and bulk techniques (IDMS based on TIMS). Early NWAL support for ES did not include an extensive QC program monitored by the IAEA and, due to lower sample capacity, many samples were only analyzed by only one particle or bulk laboratory, or were archived. SIMS particle analysis was only available at a few laboratories and was not used for a large portion of the particle work.

3.2.2. Current NWAL Support for ES

Current NWAL support for ES has expanded since initial implementation of ES, and now includes the following:

- Laboratories in the US, Russia, UK, France, Japan, Germany, Australia, and Finland;
- Preliminary laboratory certification work in Brazil, China, South Korea, and other States;
- "Surge" capacity which can handle high sample loads during special sampling campaigns
- A formal and robust QC program;
- A large contribution by SIMS to the particle analysis workload (about 40%) from seven laboratories (see Figure 3);
- Increased availability for special measurements including electron microscopy, AMS, and other techniques, which are often used to help investigate specific SG issues.

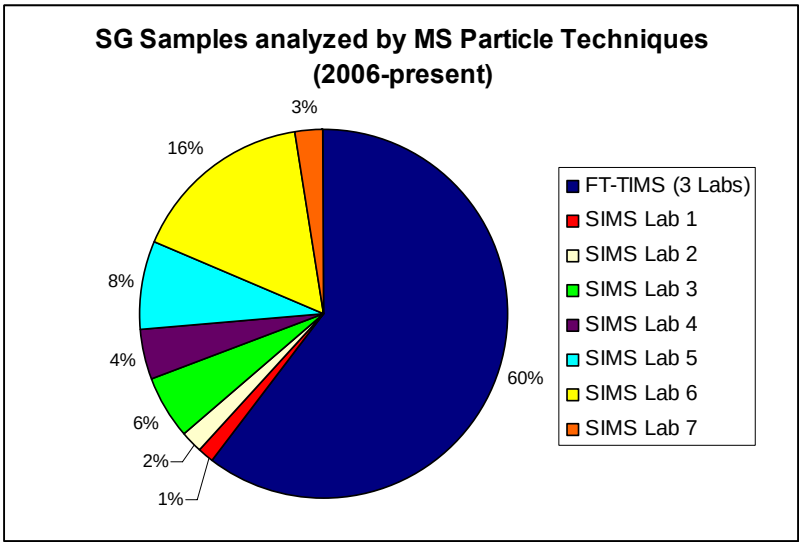

Figure 3: Breakdown of the MS Particle Technique Workload (FT-TIMS and SIMS).

3.2.3. Future NWAL Support for ES

Future NWAL support for ES is expected to expand further, with new members (such as possible participation by laboratories in Brazil, China, South Korea, and other States), new analytical techniques, and increasing sample capacity. In particular, it is expected that additional laboratories will be able to provide FT-TIMS, SIMS (conventional or UHS-SIMS), bulk analysis support, and support for special nuclear forensic analysis.

3.2.4. Overview of NWAL Support for NM Analysis

In the beginning of implementation of IAEA SG, it was recognized that DA of NM would play a major role in verifying the correctness of State declarations. Inter-laboratory exercises in the late 1960s helped to identify laboratories in Member States with the capability to accurately measure U and Pu-containing fuel cycle materials. Out of this list of laboratories a smaller number became the Agency's NWAL that was active in the 1980s and 1990s, primarily for the measurement of spent fuel input accountability tank solutions from reprocessing facilities as well as U oxide powder and pellets from fuel fabrication facilities.

Support for NM analysis at network laboratories other than SAL was highly cost-effective at times when the sample load was too high for a single laboratory to deliver results within the required timeliness criteria. However, in the late 1990s and early 2000s, the sample load of spent fuel input solutions fell dramatically and NM analysis support from laboratories other than SAL was less frequently called upon. Coincident with this drop in sample load, a number of the active network laboratories stopped the analysis of NM due to changes in regulations and business philosophy; this affected laboratories in Germany, Austria, the Russian Federation and the UK. Presently, SAL performs the analysis of nearly all NM samples, with the exception of D_2O, and other network laboratories are used to perform parallel analysis of samples for QC purposes.

The Agency is currently seeking new members of the NWAL which can analyze NM in order to strengthen this QC role and to provide back-up capacity in case of a catastrophic failure of SAL. One new network laboratory has been qualified in 2007 for NM analysis (the Institute for Trans-Uranium Elements, Karlsruhe, Germany) and two other laboratories (in France and Belgium) are under qualification.

3.3. IAEA On-Site Laboratory (OSL) at the Rokkasho Reprocessing Plant, Japan

The IAEA faces many challenges in safeguarding the Rokkasho Reprocessing Plant (RRP) in Japan because of the high throughput of NM (estimated 8000 kg of Pu per year). At the time that construction of RRP began in the early 2000s, it was decided to build a special on-site laboratory for IAEA joint use with the Nuclear Materials Control Center (NMCC) operated by the Japanese Safeguards Authority (ref. [1] and [2]). The main justification for this facility was to meet the timeliness criteria for IAEA SG on direct-use material as well as to reduce the cost and delays of shipping Pu-containing samples to the IAEA laboratory in Seibersdorf.

The concept of the OSL is for NMCC and IAEA personnel to make measurements in parallel on separate sub-samples of the process material. Measurements are performed with Hybrid K-Edge Densitometry and IDMS as well as other supporting methods. The results of such verification measurements are compared with those of the facility operator, Japan Nuclear Fuels Limited (JNFL).

The on-site laboratory approach was adopted earlier by the European Commission (Euratom) at the reprocessing facilities of Sellafield, UK and La Hague, France. Thus, a large amount of operational experience has been gained by the international SG community with such specialized on-site laboratories.

4. Areas of Growth

The move to more investigative inspections and sampling activities is leading the IAEA into areas requiring new SG analytical services and SG evaluation approaches. These services and approaches include trace-level impurity analysis of bulk U compound samples, implementation of UHS-SIMS, and support for special studies involving nuclear forensic analysis.

4.1. Trace-level Impurity Analysis of U Compound Samples

Bulk samples of U compounds are routinely taken during inspection activities (i.e. uranium ore concentrate (UOC) or "yellowcake", UF compounds, U nitrate solutions, etc.). In the past normally only U concentration and U isotope abundances were measured; elemental impurity analysis (primarily by XRF) was rarely performed. Currently, increasing emphasis is being placed on analysis

of trace-level elemental impurities by ICP-MS, a technique which generally has much lower detection limits than bulk XRF techniques.

The impurity levels can be used in conjunction with the U isotopics and other analytes to produce multi-variate analysis data to compare samples of unknown origin to increasingly comprehensive databases. When DU, LEU, or HEU materials are sampled, trace-level impurity analysis may also complement the isotopic information, for reasons such as matching to materials databases for source identification or understanding possible sample history, etc.

The most important application however is likely to be in areas such as safeguarding uranium conversion facilities, where various U compounds are involved having natural U isotopic composition; the use of trace-level impurities may allow the IAEA to obtain key information including the following:

- Characterization of imported and domestic UOC materials currently in the inventory;
- Verification of future declared receipts of UOC from imported or domestic sources, and possible detection of undeclared UOC inputs into the process (an example is given in Figure 4);
- Characterization of impurity levels in U compounds at various stages of the process (i.e. ammonium uranyl tricarbonate, UO_2, UF_4, and UF_6) using the declared and characterized UOC feed materials;
- Detection of changes in impurity levels in intermediate and final products which may indicate changes in UOC feed materials and/or changes in the process;
- Indication of at what point U materials are pure enough (i.e. such as in relation to various processing standards for elemental purity) so that application of SG may be considered; and
- Indications of changes in impurity content may also serve as input for material balance evaluators and other analysts, possibly alerting them to unusual activities and/or providing reasons to re-sample or investigate certain items, batches, or process lines.

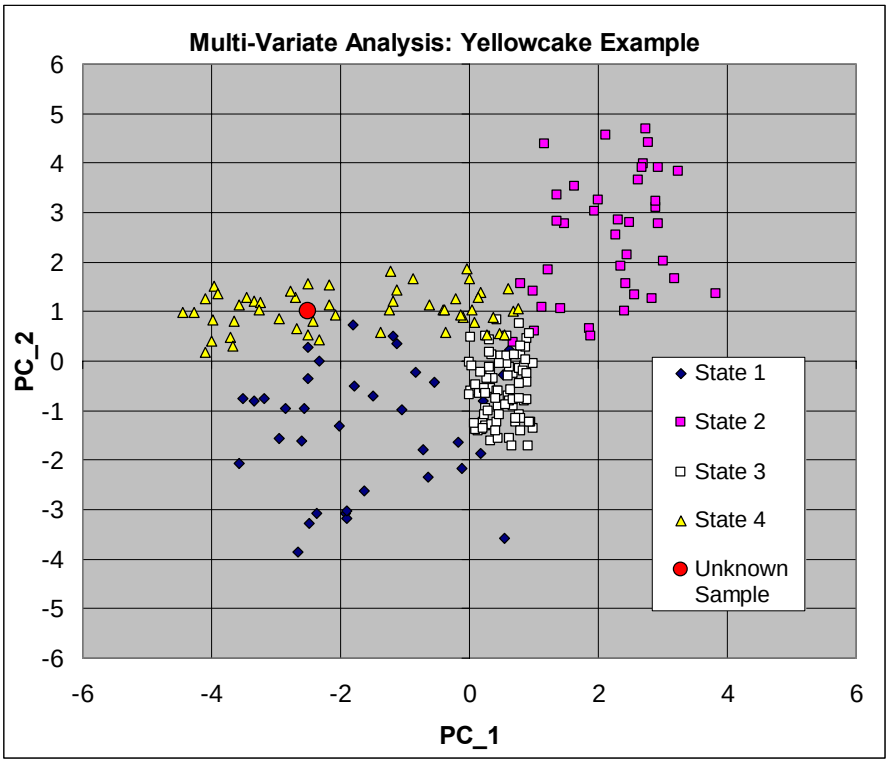

Figure 4: Example of the use of Multi-variate Analysis to characterize U materials from various sources based on trace-level impurity content. The variables PC_1 and PC_2 are linear combinations of various elemental impurity levels.

The use of multi-variate analysis and established U compound databases (containing measurement results for samples from different sources, States, etc.), makes it possible to perform evaluation functions such as calculating the probability that an unknown sample comes from a particular source.

4.2. Implementation of UHS-SIMS

As discussed earlier, conventional SIMS (such as the Cameca 4F instrument used at several of the network laboratories, see Figure 5) has played an increasingly important role in the ES program.

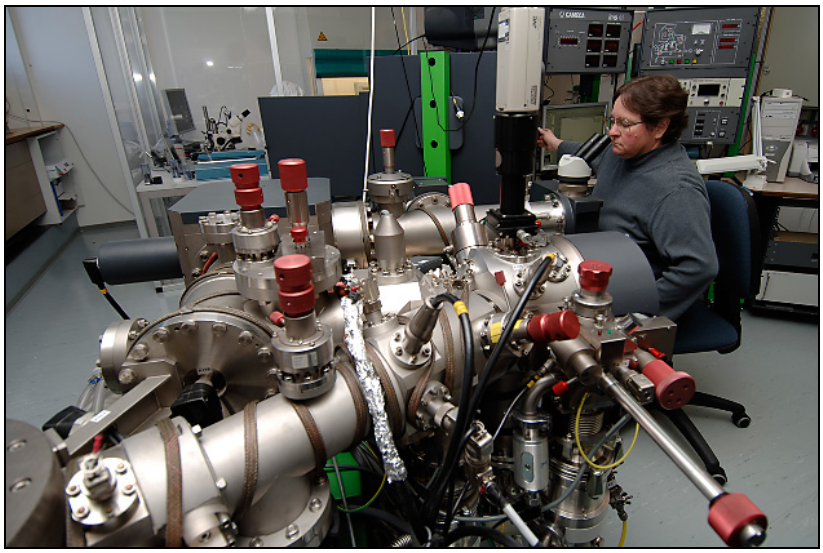

Figure 5: The conventional SIMS (Cameca 4F Instrument) currently used at SAL/CL (photo courtesy of SAL/CL).

Some advantages of conventional SIMS include simplified sample preparation, capability of analyzing a large number of particles from a particular sample, elemental analysis capabilities, etc. Nevertheless, measurement of minor isotopes of U is very challenging, primarily due to molecular ion interferences. This often results in inflated minor isotope results which cannot be used for evaluation purposes (see Figure 6). In such cases, evaluators often cannot make judgements concerning consistency with specific declarations, such as U feed material, possible sources of U materials, prior history of equipment, etc.

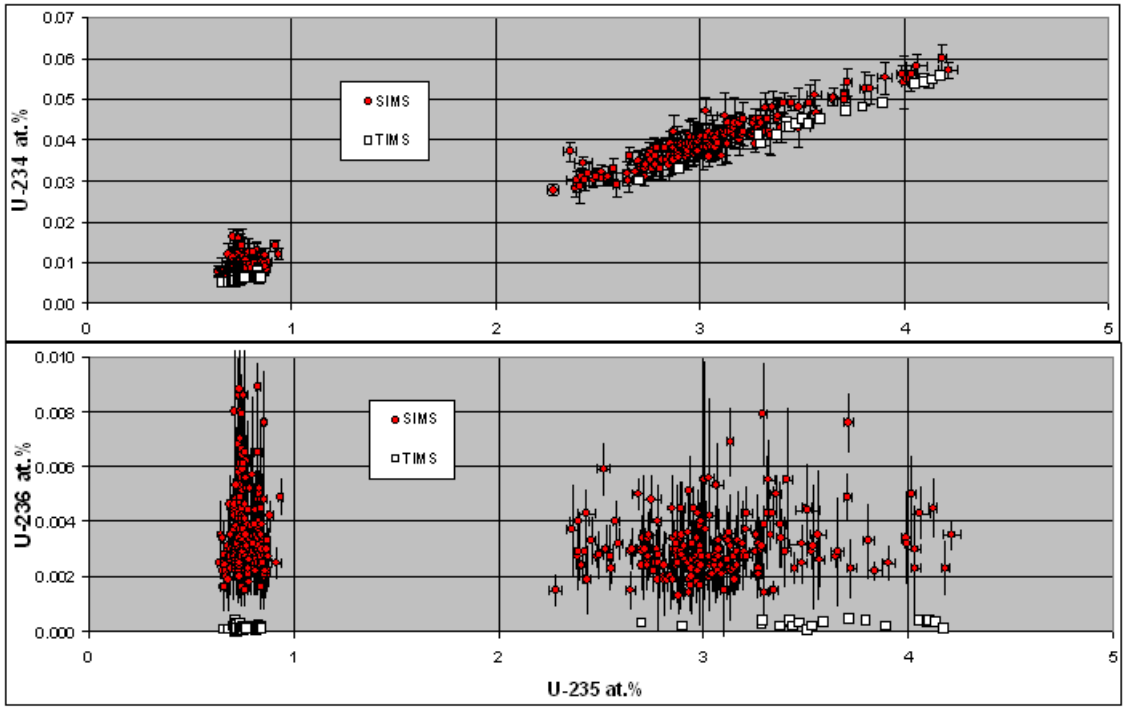

Figure 6: Comparison of TIMS and conventional SIMS particle results for a swipe containing NU, LEU, and environmental dust. This figure illustrates the difficulty that conventional SIMS instruments face when characterizing U minor isotope abundances.

These molecular ion interferences are ions with masses close to those of ^{234}U and ^{236}U, and require higher mass resolution to resolve than the Cameca 4F can offer. Ranebo et al. [3] investigated this issue very carefully and in their publication present a list of potential interferences for swipe samples and the mass resolving power necessary to avoid them. In their work, Ranebo et al. indicate that the 1270 Cameca UHS-SIMS instrument used reached a level of mass resolution that allowed for resolving of all common molecular ions except for uranium hydride interferences, including $^{235}U^{1}H$ which affects ^{236}U measurement results. The potential interferences listed by Ranebo et al. for ^{234}U and ^{236}U are listed in Table 5. It is noted that although uranium hydride interferences are still a problem for UHS-SIMS, corrections can be applied to the ^{236}U measurements by measuring the $^{238}U^{1}H/^{238}U$ ratio.

U Isotope	Interference	Mass Resolving Power (M/ΔM)
^{234}U	$^{208}Pb^{26}Mg$	2864
	$^{207}Pb^{27}Al$	2802
	$^{206}Pb^{28}Si$	2613
	$^{92}Mo^{94}Mo^{16}O_3$	958
	$^{138}Ba^{32}S^{16}O_4$	1272
	$^{116}Sn^{118}Sn$	985
	$^{48}Ti2^{138}Ba$	975
^{236}U	$^{208}Pb^{28}Si$	2566
	$^{92}Mo^{96}Mo^{16}O_3$	947
	$^{118}Sn_2$	974
	$^{116}Sn^{120}Sn$	977
	$^{48}Ti^{50}Ti^{138}Ba$	953
	$^{182}W^{54}Fe$	1496
	$^{1}H^{235}U$	**38152**

Table 5: Potential Interferences for ^{234}U and ^{236}U in Swipe Samples (from Ranebo et al.). It is noted that all common interferences in this table can be resolved with UHS-SIMS, with the exception of $^{1}H^{235}U$

Ranebo et al. also indicate additional likely improvements which UHS-SIMS offers including:

- A multi-detector counting system which allows for more counts to be collected, reducing uncertainty and likely allowing for the analysis of smaller particles; and
- Fewer problems with sample loading and faster, more effective particle searching (since more particles can be loaded onto a substrate).

The IAEA (with the assistance of the NWAL and contributions from various Member State Support Programmes such as the EC and UK) has investigated UHS-SIMS and found it to be a very useful and a logical next step in the development of its particle analysis capabilities at SAL/CL. Several of the ES network laboratories are also seriously considering implementation of UHS-SIMS, which would also benefit the IAEA. A Cameca 1280 UHS-SIMS instrument is shown in Figure 6. As previously discussed in this paper, implementation of UHS-SIMS at SAL/CL is planned for 2011.

Figure 6: The Cameca 1280 UHS-SIMS Instrument (photo courtesy of Cameca).

4.3. Special Studies involving Nuclear Forensics

IAEA inspection divisions are increasingly faced with non-standard questions where elemental and chemical characterization of individual particles may provide the answers. These samples may be swipes collected from surfaces of equipment or environmental samples such as soil or dirt, which may contain particles of nuclear materials or materials with nuclear applications. These investigations are often time-consuming, expensive, requiring state of the art instruments, and highly trained analytical staff.

4.3.1. Sample Processing

After receipt of such samples, whether they are swipes, soil, or other media, laboratories may approach sample processing in a variety of ways. For example, samples may be examined under an optical microscope to search for particles fitting a certain description (such as size, shape, color, refractive index, etc.) which may then be directly removed from the sample. Alternatively, material such as soil may be filtered or sieved by size, or separated by density and magnetic properties. Once particles of interest are found, they may be characterized by optical microscopy to help determine their possible history or composition, and photographed for further reference. Interesting particles may be extracted and loaded onto substrates for further analysis by various instruments. Some examples of SEM images are shown in Figure 7. In any case, these types of samples may contain very few particles of interest that are mixed in a sea of background particles such as naturally-occurring minerals, turning this process into a challenging one in terms of analysis time and laboratory resources.

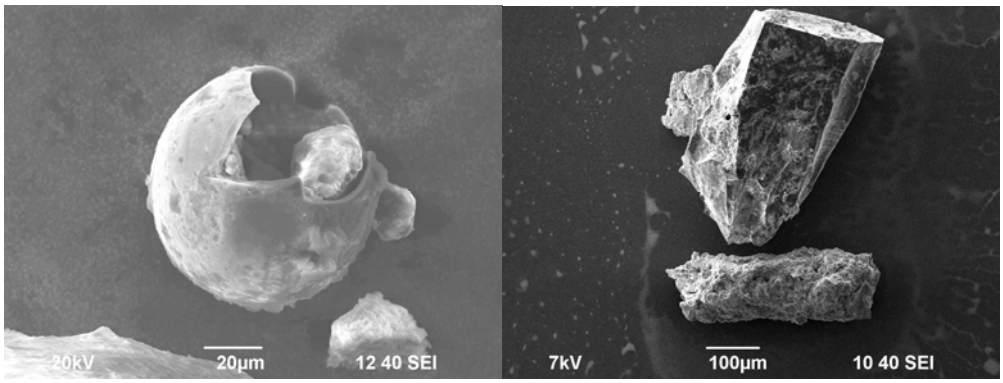

Figure 7: Examples of Particles examined by SEM (images courtesy of SAL/CL).

4.3.2. Analytical Techniques

Analytical techniques used to analyze special nuclear forensic samples may include:

- Electron microscopy - to find interesting particles and characterize their morphology, elemental and/or chemical composition;
- SIMS - to characterize isotopic and elemental composition of particles;
- Raman spectroscopy - to identify particles made of graphite or certain chemical compounds; and
- Liquid chromatography - to detect various types of high explosive materials.

These and other techniques may be employed to help answer questions related to:

- The purity of any U particles (i.e. if it is of environmental or man-modified origin);
- The presence of metals or materials used in industrial and/or nuclear applications (such as stainless steels, neutron absorbers, structural materials, etc.);
- The presence of high explosive materials which may have nuclear weapon applications; and
- Declarations concerning the separation of non-nuclear elements (such as separation of certain stable isotopes for industrial purposes, using centrifuges or laser isotope separation techniques).

5. Conclusions

The challenges facing the IAEA regarding its safeguards verification activities have changed over time and the IAEA Safeguards System has evolved to meet these challenges. In response to these challenges, the types of inspection measures applied, the types of samples taken, the laboratories and analytical techniques employed, and the types of evaluations performed, have, inter alia, been developed and improved during the past several decades. As new inspection approaches and sampling, analytical, and evaluation techniques become available in the future, the IAEA will strive to implement those which improve its detection capabilities and help provide more robust SG activities and conclusions.

5.1. Broadening of SG Verification Measures

A major milestone in the evolution of SG analytical services was the broadening of SG measures to include not only the verification of declared nuclear materials, but also the detection of possible undeclared nuclear materials and activities. While previously (in the 1970s and 1980s) sampling primarily focused on DA samples taken for verification of declared inventories and nuclear material balance purposes, the change required implementation of new sampling and analytical techniques in the 1990s, as well as new evaluation tools and approaches, which could detect and characterize trace-level amounts of nuclear materials and signatures important in nuclear processes. Detection of these small signatures may alert the IAEA to possible undeclared materials and activities.

5.2. Laboratory Infrastructure

Since the 1970s, when SAL was first built, the IAEA has expanded and adapted its laboratory infrastructure, including new sample processing areas and analytical instruments, to accomplish the types of analysis required and numbers of samples collected. In particular, in 1996, SAL/CL was completed, which has become the focal point for the management of ES samples and initial sample processing, and maintains a variety of crucial analytical techniques employed in the ES area. In addition, various laboratories around the world have joined the IAEA's NWAL, which has the goal of providing high quality analytical services, diverse analytical techniques, and increased capacity, allowing for analysis of ES by at least two laboratories.

Currently, the IAEA is in the process of designing refurbished and/or new facilities to upgrade or replace its NM and ES laboratories. The ECAS project aims to accomplish these tasks and is well under way, with several major planning tasks already completed and with an established project timeline and budget. In particular, ECAS envisions an upgraded and modernized Nuclear Laboratory (for analysis of NM), construction of an extension to the Clean Laboratory (for analysis of environmental samples), and purchase and installation of UHS-SIMS instrument for improved U particle isotopic analysis capabilities. The ECAS project started in 2008 and is expected to complete its tasks in 2014.

5.3. Areas of Growth

Key areas of growth include:

- Trace-level impurity analysis of U compound samples - to help the IAEA assess consistency of sources and processing history of NU and other materials, where U isotopic information may not be enough to verify consistency with declarations;

- Implementation of UHS-SIMS - primarily for improved U minor isotope performance for U particles (comparable to that of FT-TIMS); and

- Special studies involving nuclear forensics - inspection samples are increasingly being taken to solve specific investigative questions requiring isotopic, elemental, and/or chemical characterization of particles, using advanced analytical techniques and requiring high levels of expertise and laboratory resources.

6. Acknowledgements

The authors would like to acknowledge the many staff members of the IAEA who have contributed to SG analytical services in the past and are currently contributing, as well as the staff at the various organizations and network laboratories which have greatly contributed to SG analytical services via sample analysis, technical advice, and cooperation with the IAEA.

7 Legal matters

7.1. Privacy regulations and protection of personal data

The authors agree that ESARDA may print this article in the ESARDA Symposium proceedings or any other ESARDA publications and when necessary for any other purposes connected with ESARDA activities.

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SESSION 9

NON DESTRUCTIVE ASSAY – ADVANCED GAMMA & NEUTRONS

FLOW-TYPE HPGe DETECTOR WITH THROUGH CHANNEL

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Abstract:

The flowing HPGe gamma-radiation detector with the through channel is intended for control of the fresh nuclear fuel elements, transferred through the detector, as well as for on-line control of the fluids and gases flows with low activity.
The p-type HPGe crystal, which generally applied for the manufacture of the standard coaxial detectors with registration efficiency 10%, was used for the flowing detector manufacturing. The central through hole was made by the axis of the coaxial crystal with two open ends. The cryostat of the detector has the cover of a special design with the through channel of diameter 10 mm which comes via the through channel in the crystal. Thus, coming through the channel in the cover, made as the aluminum tube, radioactive sample (fuel element, fluids or gases flow) is found inside the germanium crystal and the registration geometry comes close to 4π-geometry.
The experimental curves of the registration efficiency of the developed flowing and standard coaxial detector of the similar volume are presented. The flowing detector has efficiency registration of gamma quanta with energy 200 keV 10 times higher, with energy 80 keV 20 times higher but with energy 40 keV – 70 times higher. At the same time the lower limit of the energy range for the developed flowing detector was 20 keV compare to 40 keV for the standard coaxial detectors based on HPGe p-type crystals.
To demonstrate high efficiency of developed detector, the spectra of various radionuclide sources with very low activity additive of Th-232 are presented. It is shown that energy resolution and peak shape in the spectra provide the precision analysis of radionuclide composition in reactor materials and low active flows of the fluids and gases.

Keywords*:* radionuclide analysis

1 . Introduction

The control for the equitability distribution of uranium and/or plutonium in mixed powders and tablets, which are put in the fuel rods is required to avoid the getting of the tablets with various enrichment degree when fresh nuclear fuel is made for the nuclear reactors [1,2]. The same control is necessary for the ready fuel rods at the final quality inspection. It is known [2], that activity of fuel rods is rather low (several hundred quanta per second) at 2÷4% enrichment. As a rule the measurement time is up to 8 hours and that is why the possibility of express measurements is an acute item. To increase the express mode of the measurement the multi crystal detectors or the detectors with the geometry close to 4π are required. The measurement time and rate

of the fuel rods shift are selected due to the sufficiency of the statistics in peaks to provide the necessary accuracy. ,
As the rule, at the control of the enrichment degree the methods to define the ratio of the intencity of uranium X-ray radiation lines to the intencity of uranium-235 lines are used [1,2]. With the account of the presence of the additional peaks with energies 90÷200 keV from the interfering impurities in the spectrum area, the application of the high-resolution HPGe detectors in many cases improves the accuracy of the required measurements.
High sensitive gamma-radiation detectors with high energy resolution are required not only on the final operations of the fresh fuel control but also at the intermediate stages of the inductrial cycle for the technological control [3]:
- gaseous and liquid uranium fluorides in the process of its production, enrichment and final control;
- plutonium solution on the various stages of its breeding;
- mixture of plutonium and uranium isotopes in separating industry;
- liquid low active wastes on the objects of nuclear fuel cycle;
- gas emissions on the nuclear energetic enterprises;

The present paper presents the results of the development of flow-type gamma-radiation detector with through channel on the basis of coaxial HPGe crystals with two open faces. Such detector could be applied for the continuous automated control of radionuclide composition and gamma activity of liquid and gas products as well as the activity of the fresh nuclear fuel rods, shifting via though measurement channel in the detector.

2. Detector design

HPGe p-type crystal of diameter 43 mm and height 40 mm, usually applied for the manufacture of standard coaxial detectors GCD-10175 with registration efficiency 10% [4], was used this time for the fabrication of flow-type detectors. The central through hole was made by the axis of the crystal. n+ contact with thickness of about 0.7 mm was made on the outer generating surface of the cylinder by the lithium diffusion. P+ contact was created on the inner generating surface by metallization of the several microns thickness.

Fig. 1. Flow-type HPGe detector with through channel

The detector in a special holder, where the input stage of preamplifier is also placed, is located in a vacuum cryostat, cooled by the liquid nitrogen. The cryostat, shown in Fig.1, has a special cover with through channel of 10 mm diameter, made from aluminum or beryllium tube. This tube goes through the hole in the germanium crystal holder. Thus, radioactive sample, going inside the cover channel, appears inside HPGe crystal and irradiation geometry is approaching to 4π.

3. Characteristics of the Detector and Their Discussion

Volt-ampere and capacity-voltage characteristics of the detector were measured at the temperature T = -185C. Depletion voltage of the detector turned out to be equal to 1000 V. The capacitance of the detector was 52 pF, what exceeds in 2 times the capacitance of the ordinary coaxial detector GCD-10175 with registration efficiency 10%. Leakage current is equal to 50 pA and 170 pA at 1000V and 2000V accordingly. Energy resolution of the developed flow-type detector at the optimal voltage 1300 V and optimal shaping time constant of 6 μs was 1140 eV and 1900 eV for the energies 122 keV and 1332 keV, accordingly. The standard detection unit GCD-10175 with the comparable sizes of germanium crystal has energy resolution 825 eV and 1750 eV for the energies 122 keV and 1332 keV, accordingly [4]. It is obvious, that the deterioration of energy resolution at the flow-type detector in comparison to the standard coaxial detector is caused by the increase of the own capacitance of the crystal as well as by the constructional capacitance of the measurement through channel.

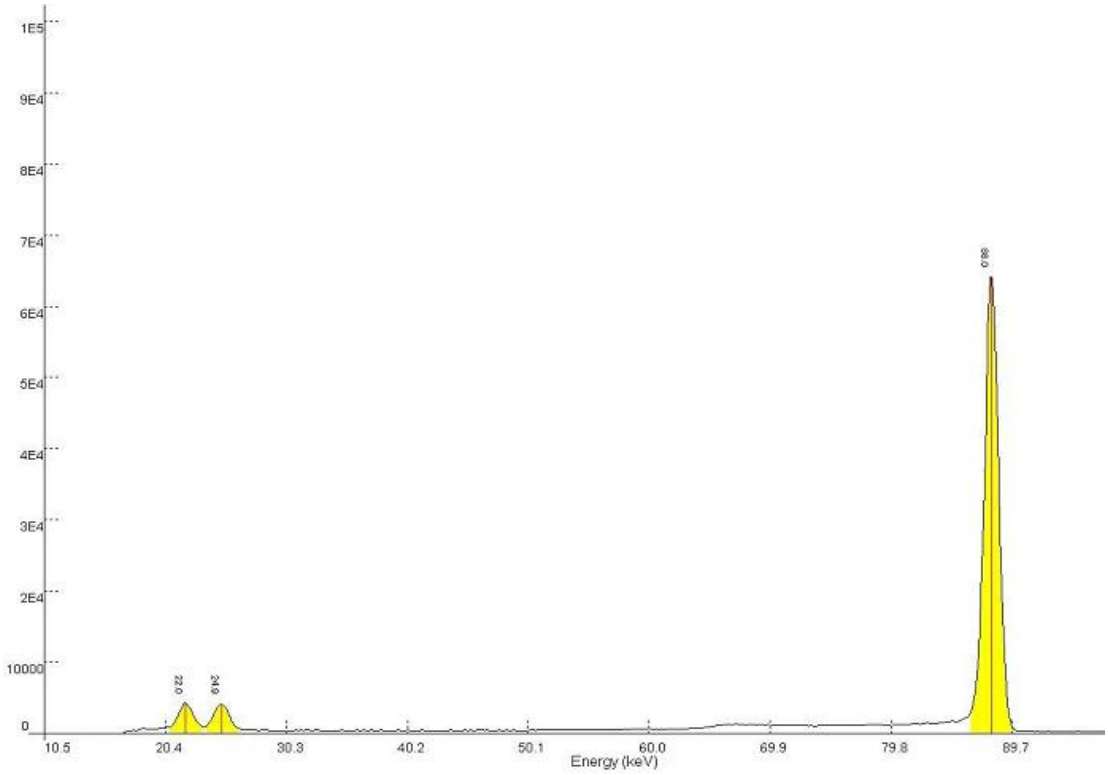

Fig. 2. Radiation spectrum of the radionuclide Cd-109

The advantages of the through detector are appeared only at the radiation on the inside at the measurement in 4π geometry. Fig. 2 shows the radiation spectrum of the radionuclide Cd-109, where the peaks 22,1 keV (AgKα), 24,9 keV (AgK_β) and 88 keV are shown. Thus, the lowest limit of the energy range for the flow type detector is equal to 20 keV at least compare to 40 keV at the standard detectors of GCD type.
Fig.3,a shows the radiation spectrum from the tungsten sample with the addition of thorium of low activity, but the Fig. 3,b – its low energy part of this spectrum. Numerous peaks of K_x series of tungsten and thorium are seen.

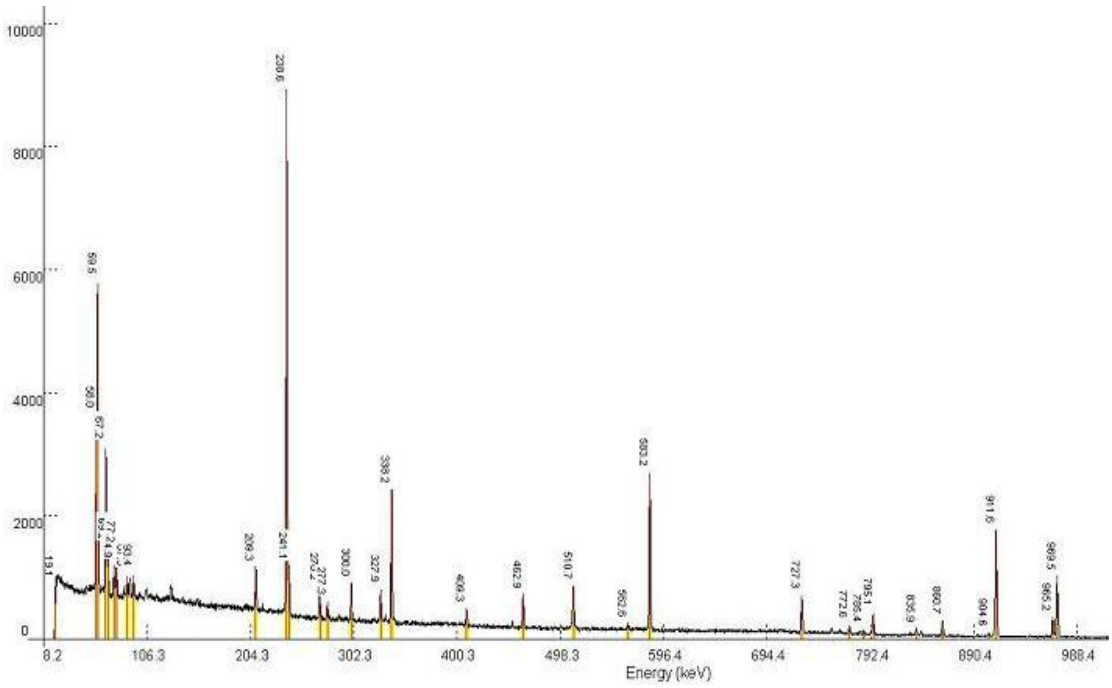

Fig.3,a. Radiation spectrum of tungsten sample with addition of low activity thorium

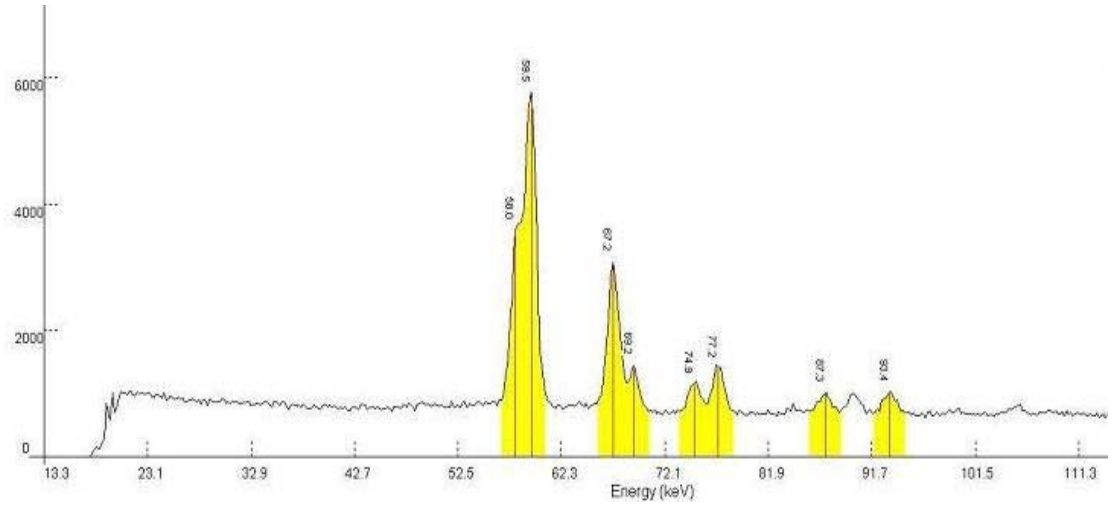

Fig.3,b. Low energy part of the radiation spectrum of tungsten with addition of low activity thorium.

Fig. 4,a shows the radiation spectrum from the source, containing the isotopes Cs-137, Co-60, Eu-154 and Eu-155 with radiation energies 32,06 keV (BaKα), 36,35 keV (BaKα), 123 keV (Eu-154), 661 keV (Cs-137), 1173 keV (Co-60) and 1332 keV (Co-60), but the Fig. 4,b – its low energy area of this spectrum.

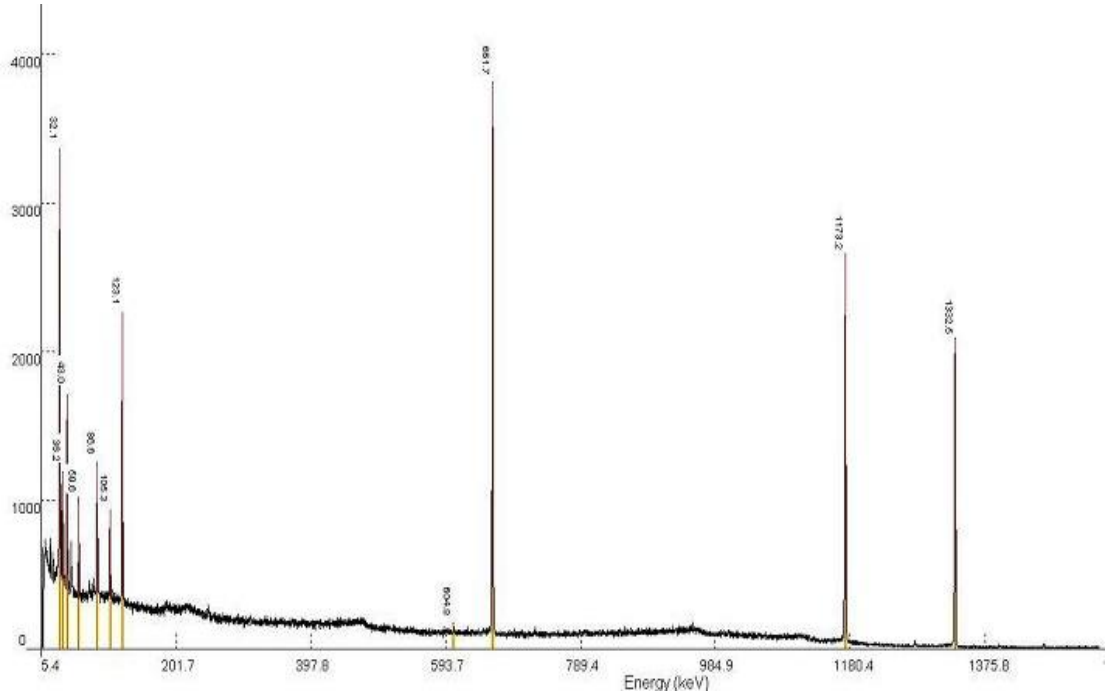

Fig.4,a. Spectrum of the sample with isotopes Eu-154, Eu-155, Cs-137 and Co-60 (common)

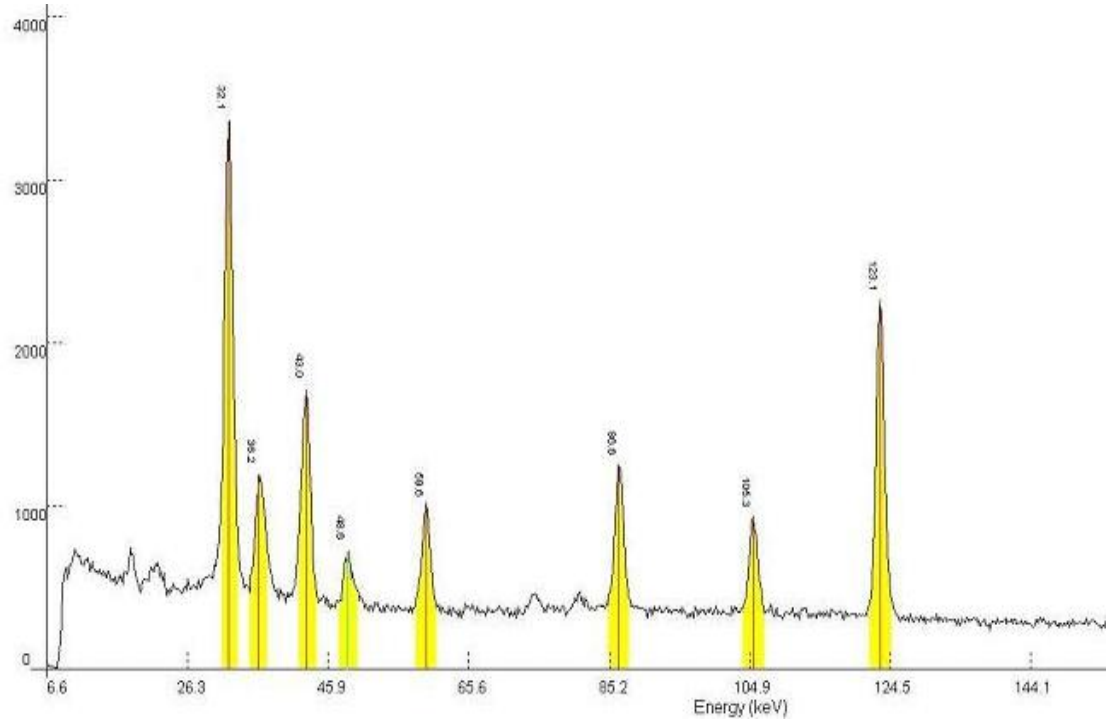

Fig. 4,b. Spectrum of the sample with isotopes Eu-154, Eu-155, Cs-137 and Co-60 in low energies range

As Fig.2-4 shows, the energy resolution and peaks shape in the spectra are rather satisfactory for the quantitative analysis of the reactor materials composition. The comparison results of the efficiency of gamma radiation registration of the flow type detector at the placement of the source in the measurement channel and coaxial detector of standard design with the same sizes of the detector crystal (10% efficiency) are interesting. Fig.5 shows the experimental curves of the registration efficiency in both cases in energy range of 20 to 1332 keV. It is seen that the registration efficiency of the flow-type detectors is considerably higher at all energies. The registration efficiency of gamma quanta with energy 200 keV at the flow-type detector is higher in 10 times, with energy 80 keV in 20 times and with energy 40 keV – in 70 times.

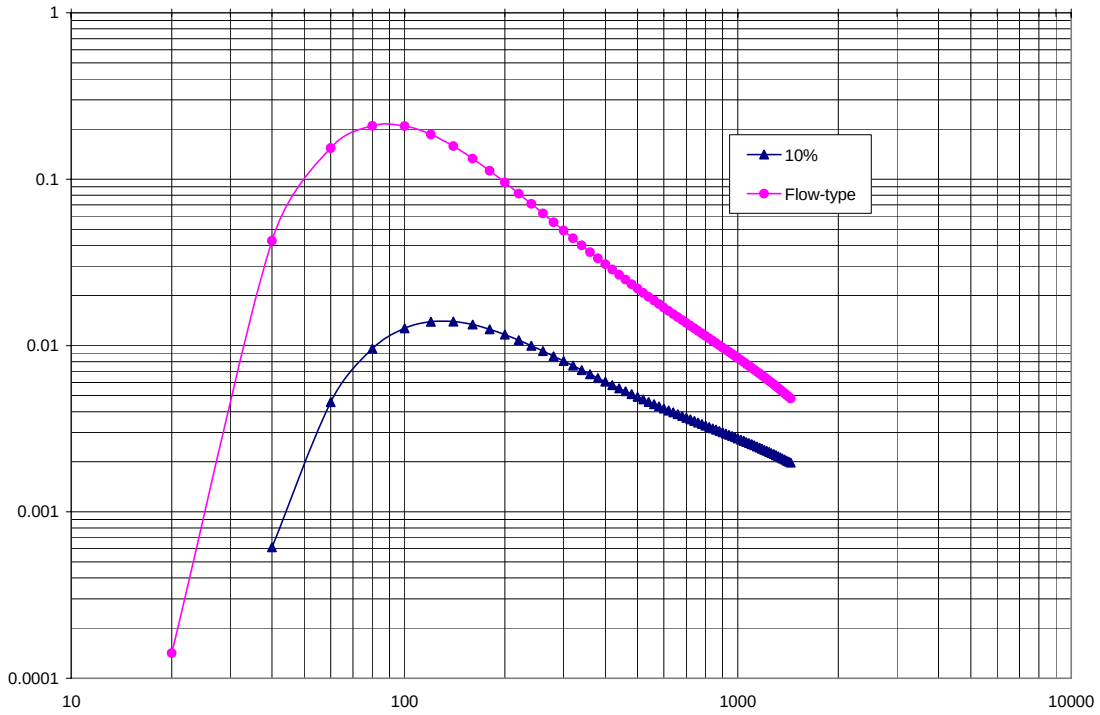

Fig.5. Comparison of the registration efficiency of coaxial and flow-type detectors

As it was mentioned above, HPGe p-type crystal of diameter 43 mm and height 40 mm which is usually applied for the manufacture of standard coaxial detectors GCD-10175 with registration efficiency 10% was used for the fabrication of the present flow-type detector. The realization of 4π geometry for such crystal has provided the multiple increase of the gamma radiation registration efficiency without the growth in volume of the HPGe crystal itself. It is obvious, that the detector of this design could be realized also on the crystals with efficiency 100 % and more. It will increase manifold additionally its registration efficiency and will decrease the measurement time for low active products to provide high energy resolution.

The application of the lead shield for the screening of flow HPGe detector against the outer radiation could also reduce considerably the lower limit of the radionuclide detection and will raise the accuracy of the measurements.

At this the automated on line control system on the basis of flow type HPGe detector could be organized the same way as it was done by us in automated monitoring system of the first contour coolant of nuclear reactor based of the standard coaxial HPGe detector with efficiency 10% [5]. Such automated control system easily could be included into the automated system of radiation control at the enterprise.

4. Conclusion

The present work presents the development results of high effective flow type HPGe detector with through measurement channel which provide the measurement geometry close to 4π geometry. Comparative measurements of the developed detector with the standard coaxial detector of the same volume have demonstrated that the gamma quanta registration efficiency with energy 200 keV in flow type detector is higher in 10 times, with energy 80 keV in 20 times and with energy of 40 keV in 70 times. Besides, the lower limit of the energy registration for the flow type detector was lower than 20 keV and at the same time the standard coaxial detectors it was equal to 40 keV.

Such characteristics enable to apply the developed detector, for example, for the nuclear fuel rods characteristics control, which could be shifted in through detector channel. The other application of the developed detector could be on-line monitoring of the low active liquids and gases which pass by though detector channel at radiochemical industry.

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A methodology for evaluating the performance of software, using γ spectrometry, for determining the isotopic composition of plutonium

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Abstract:

This paper presents the progress of an ongoing study within Working Group N°27 of the CETAMA[1] regarding the performances of software used for determining the isotopic composition of plutonium and uranium, by means of γ spectrometry, in the presence of "disturbing" radioactive emitters and of various matrices. The "disturbing" radio-emitters are some minor actinides (242Am, 243Am, 237Np, 243Cm etc.), fission products (137Cs, 125Sb, 154Eu etc.) and/or activation products (60Co etc.). All these radionuclides can be found in waste from the nuclear industry. Matrices can also vary (metal, vinyl…). In some instances, these radionuclides and matrices are in high enough quantities to disturb the analysis by the software for determining the isotopic composition. This paper presents the outline and first achievements of this .

[1] The CETAMA (Commission d'Etablissement des Méthodes d'Analyse), part of the CEA, has served laboratories for analysis and monitoring facilities throughout the French nuclear sector. Working group N°27 brings together experts in neutron and gamma spectrometry measurements to identify suitable non destructive assay methods for waste packages of all shapes, all backgrounds and all levels of alpha or beta-gamma emitters.

Keywords: waste; isotopic composition; γ spectrometry; interference

1. Introduction

The French organisation CETAMA (Commission d'Etablissement des Méthodes d'Analyse) must fulfil several missions for its customers in the nuclear industry:

- To propose scientific and technical developments aimed at improving the quality of chemical or nuclear measurements
- To collect and harmonize analytical and measurement procedures
- To organise inter-laboratory comparisons

The CETAMA is organised around Working Groups aimed at facilitating the identification of needs, sharing lessons learned and disseminating knowledge in the field of analysis and measurement. The study described in this paper has been initiated and managed by the "*Measurement of VLA/LA and MA packages*" sub-group within the "*Non destructive analysis for waste and decommissioning*" Working Group (WG 27).

Isotopic composition (IC) determinations based on gamma measurements are commonly used to characterize nuclear waste when the physical (geometry, density) and radiological characteristics of

the packages allow using the part of the gamma spectra situated between 50 and 200 keV. When the conditions are favourable, in most cases, gamma spectrometry can be used to quantify the activity of one ore several actinides by using their emission lines between 150 and 600 keV.

An inter-laboratory round robin has been organised by the CETAMA from 2002 to 2004, during which five drums of simulated and actual solid waste were circulated between the 10 participating laboratories for the determination of activity for beta-gamma emitters, the determination of isotopic composition for U or Pu and the determination of activity and weight for each isotope of Pu (or U). This round-robin gave rise to numerous discussions and extensive analysis work on the lessons learned. During this discussions, the group of experts determined that it was necessary to obtain more information on potential difficulties and limits of use when applying the various isotopic composition software tools in the presence of some “interfering” radionuclides (actinides, fission or activation products), liable to be present in the waste.

Numerous studies have been performed in the past to determine the influence of measurement conditions and to estimate the precision of the various codes on simple configurations (samples). The M.G.A. code, developed by Lawrence Livermore national Laboratory, is currently the most widely used software for isotopic composition determination. PC-F.RA.M (Fixed-energy Response function Analysis with Multiple efficiency), developed at Los Alamos National Laboratory) is the second most used code in this field. A team of experts at CEA Saclay is developing a new code I.G.A. Those three codes allow analysing gamma spectra recorded via germanium detectors. At the time of the study, very few information existed about the way those three codes dealt with “interfering” elements, especially when their concentration becomes significant.

A bibliographic study performed on those three codes showed that they operate on very different principles. The interfering elements of interest in common waste material are not always detected / quantified systematically, and may sometimes generate errors in the IC. This analysis thus confirms the need for a detailed and systematic analysis of the impact of these radionuclides.

2. Original specifications for the study

The original aim of the study was to determine the field of applicability of each IC determination software tool, when considering various types of package, radiological contents, measurement conditions and configuration, and equipment.

One method to perform this evaluation consists in obtaining spectra from waste packages with known isotopic compositions, and to submit these spectra to analysis by the various IC tools, in order to compare the results with the known composition.

In a first step, and for the sake of simplicity and exhaustiveness, it was proposed to generate these “test” spectra numerically. However it was soon concluded that computer simulation was not adapted, since, in the region of interest (around 100 keV), it was difficult to accurately model the interactions responsible for the continuous background in these energy ranges. One of the reasons for this is that the nuclear data used to describe these background phenomena are not well known (for absorption and diffusion mechanisms, specifically). In addition, the available nuclear and atomic data for the emission lines of actinides are sometimes known with significant uncertainties. The values found in the reference databases, and related to the peaks found in this region, differ both for the energy values and for the branching coefficients. Thus, by performing the evaluation on simulated spectra, it would not have been possible to discriminate between the actual performance of the IC software and the eventual differences in the databases used. It was then decided to perform the evaluation on spectra derived from experimental measurements.

A common project involving CEA, AREVA and IRSN was then launched, driven by the DTN.SMTM/LMN (CEA Cadarache) assisted by CETAMA. It was expected that the project would last for about 2 years.

The proposed method involved the following steps:

- Acquisition of “elementary” mono-isotopic or mono-elemental spectra on mock-up packages, with standard sources in well defined measurement conditions, to establish a common database.
- Computation of “virtual” spectra by summing these “elementary” spectra on each channel (in the same conditions).
- Testing of these virtual spectra (or TEST spectra) with the various software tools.

According to the original program, it was proposed to perform the acquisition of the “elementary” spectra on various package mock-ups, with each elementary standard source, at several positions in the package, and in conditions as reproducible as feasible.

A preliminary specification file was established to determine all the potential parameters of influence, and fix the test conditions. The preliminary matrix involved 4 to 10 Pu and U isotopic compositions, 4 to 8 interfering nuclides (actinides, fission products (FP) and activation products (AP)), and 4 packages with several possible source locations, for at least three types of detectors and varying measurement configurations. An example of a potential package configuration is presented on fig. 1.

Figure 1: potential mock-up configuration

3. Evolution of the specifications

Soon in the process, it became evident that the time required to perform all the acquisitions would not allow completion of the project in the expected time frame. This resulted from the very large number of parameters that needed to be studied to establish the database (number of sources, number of detectors, number of matrices, and configurations...). In addition, some of the sources required to explore the whole range of target “interfering” radionuclides were not available. The specifications were then adjusted to this new context.

3.1. Detectors.

The group decided not to reduce the number of detectors, since it is necessary to provide information for a wide range of users.

3.2. Test assemblies.

The major proposed change consisted in performing all the required “elementary” acquisitions on a specific measurement bench instead of mock-up packages. This would allow significant simplifications in the process:

- Elimination of the package preparation step, and of the associated logistics to combine the packages and the sources and transport them to the measurement stations.
- Decrease of spatial requirements, thus allowing the implementation of three detectors with their electronics in a single room.
- Optimization of the counting time, since the distance between the detector and the source would not be constrained in the same way.
- Improved reproducibility
- Enlarged field of applicability (not limited to the geometry and configuration of the mock-up packages).

This change was accepted provided that the equivalence between the spectra obtained on the bench and on mock-up packages is validated.
Later, and since the bench allowed it, it was decided to add another kind of matrix to the study: a mixture of 50 wt% PVC and 50 wt% glass, with densities between 0.2 and 0.6 $g.cm^{-3}$.

3.3. Radionuclides.

No sources were identified for Co-60, Eu-154, Sb- 125, Cm-243 or Am-242m. The available sources were thus the following:

- 3 Pu sources containing between 80 and 97 % Pu-239
- Mono-isotope sources (28 years old) for Pu-238, Pu-240 and Pu-241
- Sources of interfering radionuclides: Am-241, Am-243, Np-237, Cs-137 and U-235.

Note: from the available Pu sources and the mono-isotopic sources, it may be difficult to simulate LWR reactor-grade plutonium, since for some isotopes the weighting coefficients would be quite high.

4. Validation of the principles

4.1. Feasibility of operations on “elementary” spectra

The creation of virtual spectra by summing elementary spectra on each channel is theoretically feasible. The only physical limit is associated with the interactions (X-ray fluorescence) that may exist between the two sources when they are together, and that cannot be observed when measuring the sources independently. As a result, and since the IC software do not consider this type of rays, the comparison will only consider the field of gamma emissions.

In practice, the quality of the combination will depend on the stability of the measurement chain. The process must be verified experimentally. In order to validate this procedure, a preliminary experiment was performed with only one single collimated low energy Ge detector (2000 mm^2, 20 mm thickness). 4 sources of varying activities were selected, representing two known isotopic compositions. The conditions were selected to allow the comparison: effective recording times identical for all the spectra to be compared, subtraction of background for each spectrum, completely identical experimental conditions.

For direct summation of spectra, the sequence of acquisitions described in table 1 was performed, both directly and through a PVC screen representing a waste matrix, for two sources with slightly differing IC.

	Source	Active recording time	Distance between source and detector
Background 1	None	T1	:
Source 1	S1	T1	D
Background 2	None	T2 (> T1)	/
Source 2	S2	T2	D
Background 3	None	T2	/
Combined source 1	S1 and S2	T2 (T1 with both S1 and S2 and (T2-T1) with only S2)	D
Background 4	None	T2	/
Combined source 2	S1 and S2	T2 (T1 with both S1 and S2 and (T2-T1) with only S2)	D
Background 5	None	T2	/

Table 1: measurement sequence to test the feasibility of summing elementary spectra

The “net” spectra were obtained by subtracting for each channel the background from the corresponding spectrum. The “sum” was obtained by summing, for each channel, the net spectra from Source 1 and Source 2, as illustrated on fig.2. Two residuals were then calculated. The “statistical and experimental” residual was obtained by difference between the two “Combined source” spectra. The summation residual was obtained by difference between the “Combined spectrum 1” and the above-defined “sum”. The residuals were compared to each other and to the background in order to evaluate the significance of the variation between the sum and the combined spectra.

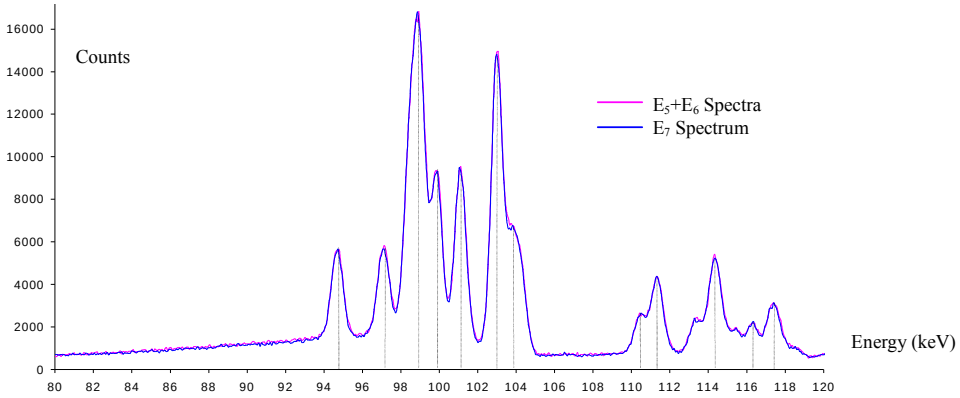

Figure 2: example of direct summation, with a PVC screen

A validation was also performed for weighted sums, to allow simulating the spectrum of a mixture of components from the elementary spectra obtained with single sources. Weighting a net spectrum (i.e. multiplying the contents of each channel by a constant) could reflect, for instance, a change in the activity of the source, or a change in the distance between the source and the detector. For the experimental study, it was decided to weight the elementary spectra from given sources according to counting time and the distance between the source and the detector. Again, the study was performed both with and without the presence of a PVC screen to simulate the waste matrix. The experimental sequence in both cases is given in table 2.

	Source	Active recording time	Distance between source the detector
Background 11	None	T5	:
Source 5	S1	T5	D1
Background 12	None	T6	/
Source 6	S2	T6	D2
Background 13	None	T7	/
Combined source 5	S1 and S2	T7	D3
Background 14	None	T8	/
Combined source 6	S1 and S2	T8	D3
Background 15	None	T8	/

Table 2: measurement sequence to test the feasibility of a weighted sum of elementary spectra

The weighting is performed by adjusting the counting time (by the ratio of the counting times) and the distance (by the square of the distance ratio) to the selected common reference values.

These experiments allowed concluding that the rebuilding of a multi-source spectrum from elementary single source spectra is possible both with and without the PVC screen, as illustrated on figure 2, for instance. The quality of the virtual spectra is nevertheless very dependant on rigorous performance of the acquisitions:

- The energy/channel relationships on the elementary spectra (and on the backgrounds) must be strictly identical. A specific preliminary tool was developed to re-compute the contents of each channel from a spectrum according to the required energy/channel relationship, by first a compression and then a linearization of the spectrum.
- The distance between the source and the detector must be large when compared to the dimensions of the source or of the detecting surface.
- The acquisition time and distance between the detector and the source must be identical, or re-normalized for weighted sums.
- Anisotropic sources must be rotated to obtain an average spectrum and avoid any fluence rate variation

4.2. Equivalence of the spectra obtained on the bench and on mock-up drums

In order to understand the effect of package configuration on the spectra, three series of tests were performed, using the same plutonium source:

- acquisition of the spectrum with the source situated in the centre of a simulated package filled with a homogeneous matrix
- acquisition of the spectrum with the source situated in the centre of a simulated package filled with a heterogeneous matrix, but with the same average density as the previous package
- acquisition of a “virtual” spectrum on the bench.

The homogeneous package was a 118 L steel drum filled with small cylinders of polystyrene coated withy PVC, in order to obtain an average density of 0.29 $g.cm^{-2}$ (see fig. 1 for instance). The heterogeneous package was obtained by inserting in the “homogeneous” matrix one block of solid PVC and empty PVC containers, so that the average density of those two insertions result in the same apparent density of 0.29 $g.cm^{-2}$. Those two “heterogeneities” were positioned along the measurement axis to maximize the impact.

The two packages were measured with and without the Pu source, in rotation, with the detector positioned at mid-height. The bench measurement was performed by positioning the source (in rotation) in front of a shielded detector (to minimize the effects from the surroundings) and through a screen composed of 0.9 mm steel (to simulate the drum) and with a thickness of PVC calculated to

provide the same attenuation as the matrix in the drum. The comparison of the two spectra from the “homogeneous” package and from the bench assembly put into light the fact that the diffusion effects in the homogeneous matrix of the package were higher than those in the PVC screen used on the bench (see fig. 3 for instance).

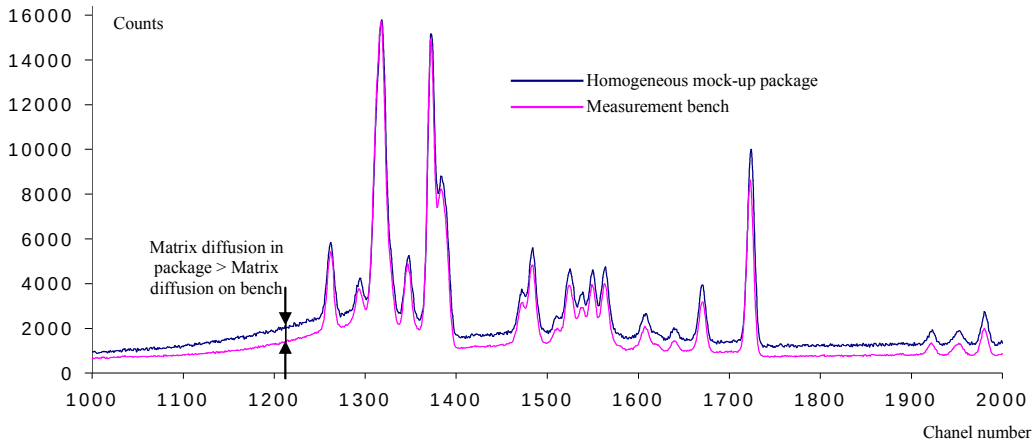

Figure 3: comparison of spectra from a homogeneous mock-up package and from the bench: effect of matrix diffusion

A systematic parametric study was then performed to adjust the bench set-up and improve the representativeness of the simulation (the aim was to obtain diffusion ratios[1] similar to those obtained with the packages). It was found that the impacts of the distance between the source and the screen or of the size of the screen were limited. The most significant improvement was obtained by adding “diffusion tunnels” to the system, between the detector and the screen and between the screen and the source, wide enough to not interfere with the detection solid angle. See fig. 4 and 5.

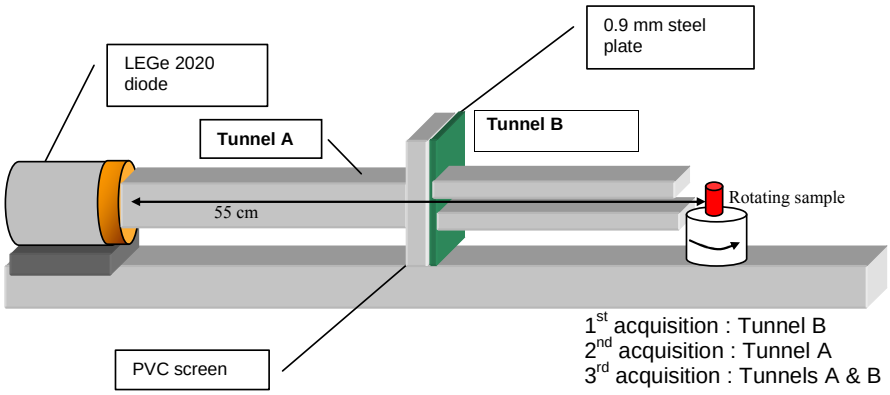

Figure 4: insertion of diffusion tunnels to improve representativeness

[1] the diffusion ratio is defined as the ratio of the area below the baseline immediately before the peak and the area below the peak, determined over equal widths of energy intervals.

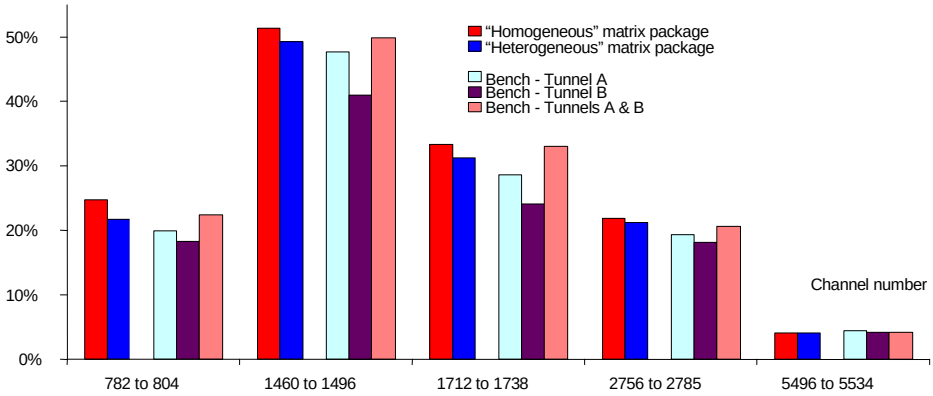

Figure 5: effect of tunnels before (A) and after (B) the screen on the diffusion ratio

5. Design and construction of the SCHEMAS bench

The SCHEMAS (*Simulateur de Colis de déchets HEterogènes pour la Mesure d'Actinides par Spectrométrie gamma*) modular bench has been designed according to the conclusions of the above studies. It has been designed to simulate the interactions between radiations and various materials that can be found in real waste drums by incorporating mobile screens with varying compositions and thicknesses, and able to simulate homogeneous as well as heterogeneous matrices. The bench is based on an existing programmable precision turntable, and can be connected to three acquisition systems at 120° from each other. See figure 6 to 8. The source is in the centre, on a rotating support. Around the source, a collimation and shielding block is positioned, to provide the three measurement windows, ensure biological shielding and limit unwanted diffusion phenomena. This block is made of "red brass" to limit unwanted X-ray fluorescence phenomena. The various matrices and their combinations are simulated by three series of 10 screens each (including steel plate, glass, Plexiglas...) that can be combined in three different zones according to the required setting. The thickness of the various screens is computed to simulate the attenuation of a reference drum, filled with homogeneous repartitions of masses, materials and source. See table 3. These 3 screen zones are positioned on the programmable turntable and separated by empty zones which allow performing the measurement of the bare source. Inside the bench, the diffusion tunnels are composed of two Plexiglas horizontal plates above and below the source.

Matrix in the drum	Average matrix density in drum ($g.cm^{-3}$)	Equivalent thickness PVC in mm (ρ = 1.19 $g.cm^{-3}$)	Equivalent thickness Fe in mm (ρ = 1.19 $g.cm^{-3}$)	Equivalent thickness glass in mm (ρ = 1.19 $g.cm^{-3}$)
100 % PVC	0.2	48.0		
	0.4	65.0		
50 wt% Fe, 50 wt% PVC	0.3	23.12	3.5	
	0.7	39.63	6.0	
50 wt% glass, 50 wt% PVC	0.3	28.0		13.1
	0.5	59.0		21.2

Table 3: Equivalent screen thicknesses to simulate various package configurations

The bench was constructed in May 2007 and set-up at the LMN at Cadarache.

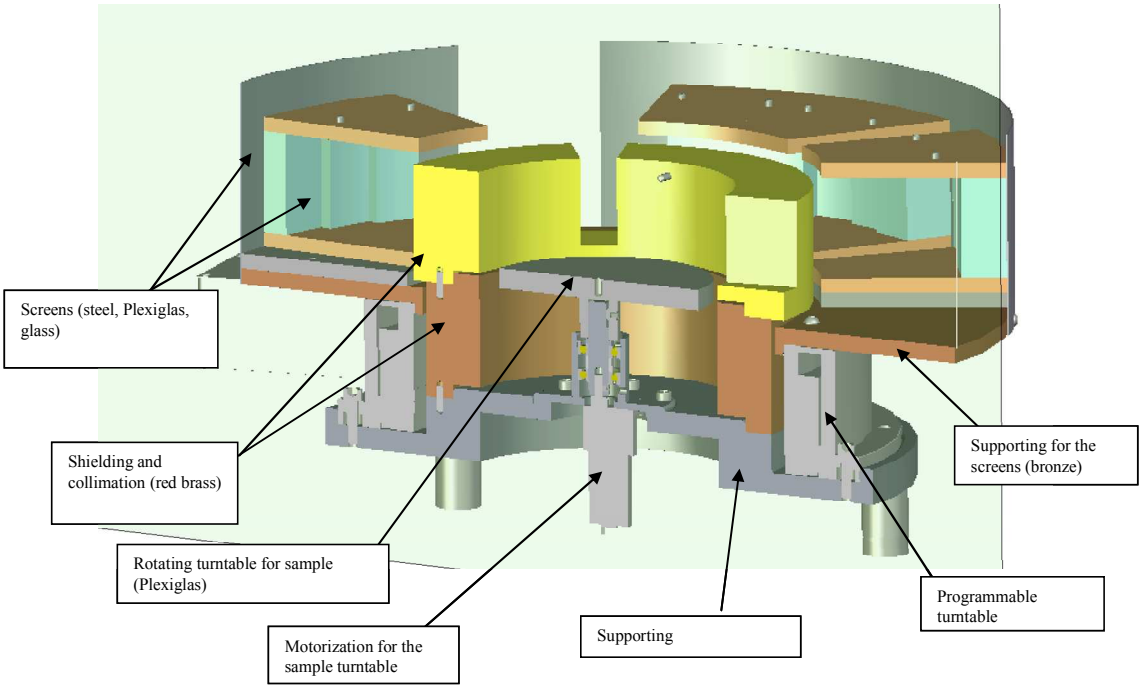

Figure 6: General view of the SCHEMAS bench

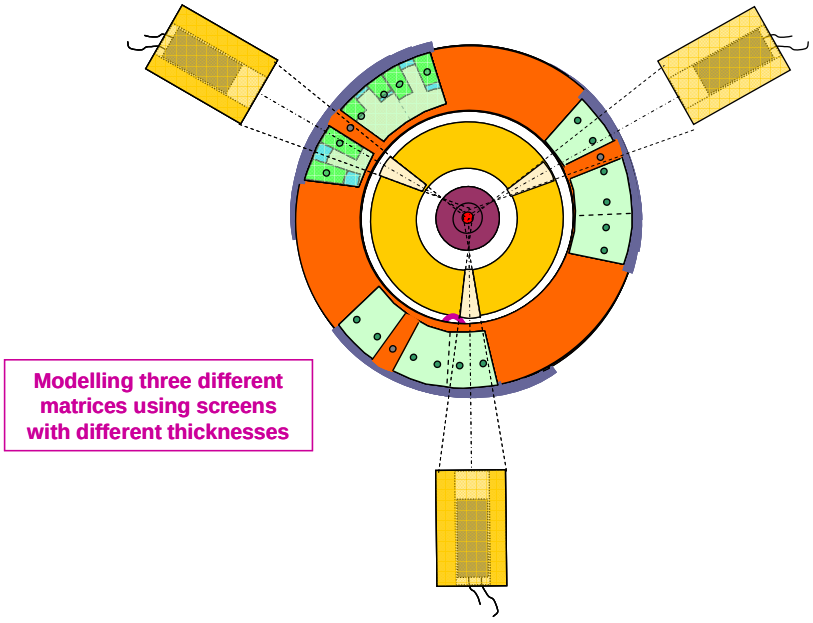

Figure 7: general disposition of the screen sequence

Figure 8: The SCHEMAS bench

6. Procedures for acquisition and creation of the test spectra

In addition to the bench tool described above, the study has allowed setting up procedures and parameters to perform the acquisitions and create the “virtual spectra” that will be analyzed by the IC software tools.

6.1. Acquisition of the elementary spectra

The criterion required by MGA to attenuate the 59.54 keV peak from Am-241 is the most conservative. It has thus been decided to apply this criterion for all the measurements. The thickness of the absorbing screen must be such that the 59.54 keV peak is attenuated to a level comparable to those present around 100 keV. The screen is made of tin (Sn).

In order to minimize the impact of any heterogeneity, the sample is systematically rotated at a constant speed. The number of rotations during the acquisition must be an integer number, or, alternatively, be so high that the effect of the non completed last rotation is not significant. The counting rate must be such that degradations associated with high counting rates (losses, piling, and decrease of the resolution) can be neglected. This is ensured by setting the distance between the source and the detector, for each source and each detector. The distances between the sample and the detector may vary between recordings, provided that the principles of weighting defined above are complied with.

The electronics should be able to be used over 16000 channels or more. The gain should be set according to the counting rate, in order to obtain an energy/channel relationship around 0.0375 keV/channel (for good counting rates) or 0.075 keV/channel (for low counting rates). If the sample contains known radionuclides that emit at least three lines spread over the required energy range, the gain is set directly with the sample. For samples that do not allow this, or for a background, a standard source must be used and three recordings must be performed (standard/sample or background/standard).

The counting time must comply with the requirements for the various software tools to be tested. The most conservative one is MGA which requires at least 10^6 counts for a plutonium sample. The criterion set for the study is 1.5×10^6 counts in the spectrum. For samples that do not contain Pu (pure interfering elements, for instance), the counting statistics associated to the various peaks must allow unambiguous interpretation.

6.2. Processing of the elementary spectra

For known spectra, the energy/channel correction is directly performed on 3 intense peaks covering the whole energy range of interest, by using the values found in a database (JEFF 3.1 for instance). The experimental data are then fit using a second order relationship and linearized using the tool described above. For spectra which require the use of a standard source before and after the measurement, it is essential to make sure that the two spectra from the standard source display the same energy/channel relationship, in order to use this relationship for the experimental spectrum. This verification is performed using three intense peaks covering the energy range of interest. The

background recordings are spectra that require the use of standard sources. They will be processed accordingly, before subtraction from the (processed) sample spectrum.

6.3. Creation of the "virtual" spectra to be tested with the IC software tools

The virtual (or TEST) spectra may be created by weighting directly the spectra according to the activities or, if the activities are not known, by indicating a proportion between the area of two peaks: a plutonium (preferably 239) peak and a peak from the pollutant. Specific attention is needed for the standardization aspects.

6.4. Compression or truncating of the spectra

The TEST spectra may then need to be compressed and/or truncated according to the requirements of the IC software tools, and converted to the adequate format.

7. Status of the study and first results

At the present date a simplified campaign is being performed, which allows testing the methodology, and the bench, and identifying the remaining integration issues, in order to finalize the software needed to operate SCHEMAS and process the acquisition results.

Acquisitions have been performed using one planar, small volume detector, with the following sources: one Pu sample with known isotopic composition (GCR-type), and three pure actinide sources (Am-243, Np-237, and Am-241). An acquisition with one Pu source and one Np237 source together has also been performed. Three types of matrices have been tested: C1 (100% Plexiglas, average density 0.3 g.cm^{-3}), C2 (50 wt% Plexiglas and 50 wt% metal - average density 0.3 g.cm^{-3}) and C3 (50 wt% Plexiglas and 50 wt% glass - average density 0.3 g.cm^{-3}). The acquisition time was between 8 and 12 hours per source. Figures 9 and 10 below illustrate the effect of the matrix on the measured spectrum and the effect of a pollutant on a recombined spectrum. Figure 9 illustrates the description by this method of the deformation (peaks and continuous baseline) of a spectrum corresponding to 10 g of GCR-type plutonium in a 200 l drum with the three different matrices (same apparent density). On figure 10, the effect of increasing amounts of Am-243 on the spectrum is computed for a C1-type package with 10 g of GCR-type Pu. It can be seen that, when the amount of Am-243 increases, additional peaks appear, with intensities that can become similar to those from the plutonium. One can also observe a change in the shape of the peaks and a progressive rising of the baseline. These effects may have an impact on the interpretation of the spectra by the IC software tools.

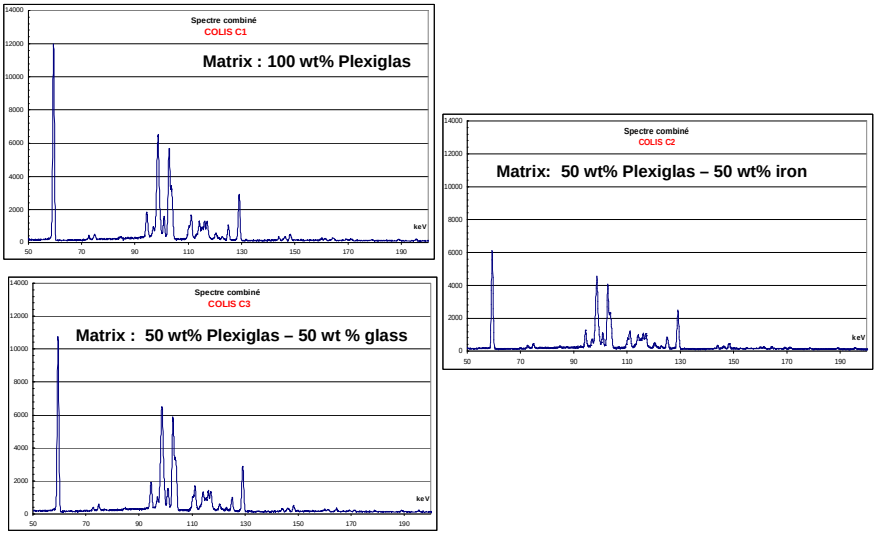

Figure 9: effect of the matrix on the "virtual" spectrum of 10 g of GCR-type Pu in 3 virtual 200 L waste drums.

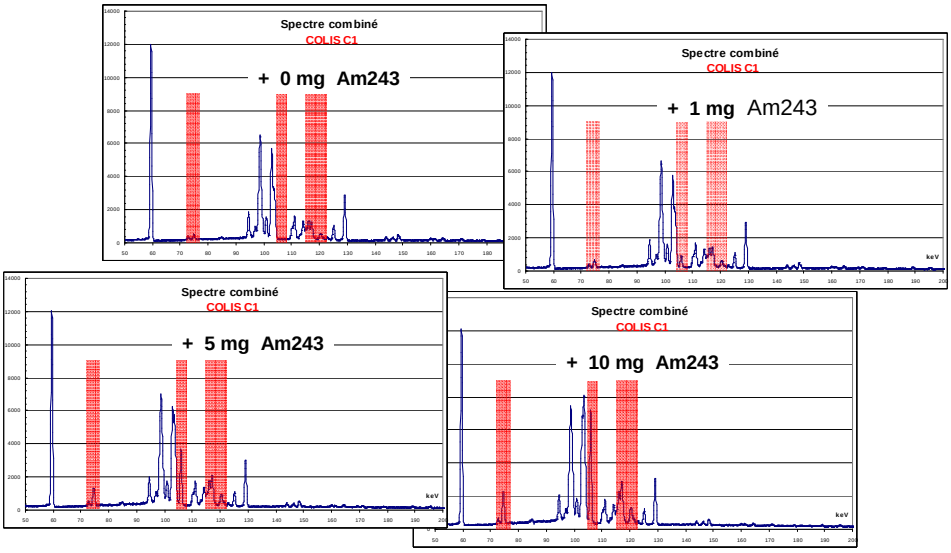

Figure 10: effect of increasing Am-243 amounts on the "virtual" spectrum of 10 g of GCR-type Pu in a 200 L drum

The raw test spectra have now been transferred to the laboratories in charge of testing and validating the method, who are now processing the spectra and subjecting them to the various IC software tools. Some minor difficulties are now being corrected: in some instances, data formatting for the various software tools under study was not consistent.

An example of the effect of these pollutants on the results given by an IC software tool is given in figure 11 below. On this figure, the ratios between the mass fractions of each isotope computed by IGA over the actual mass fraction of this isotope in the starting spectrum are plotted versus increasing Np-237 contents (which can be found in the label of each abscissa). It can be observed that, for this specific Pu IC and configuration, the addition of 0.5 % or more of Np-237 in the mix leads to serious discrepancies, especially for Pu-240 which, here, can be overestimated or underestimated by a factor of more than 2. Similar systematic testing is being performed for various isotopic compositions, interfering radionuclides and matrix compositions, using the three software tools.

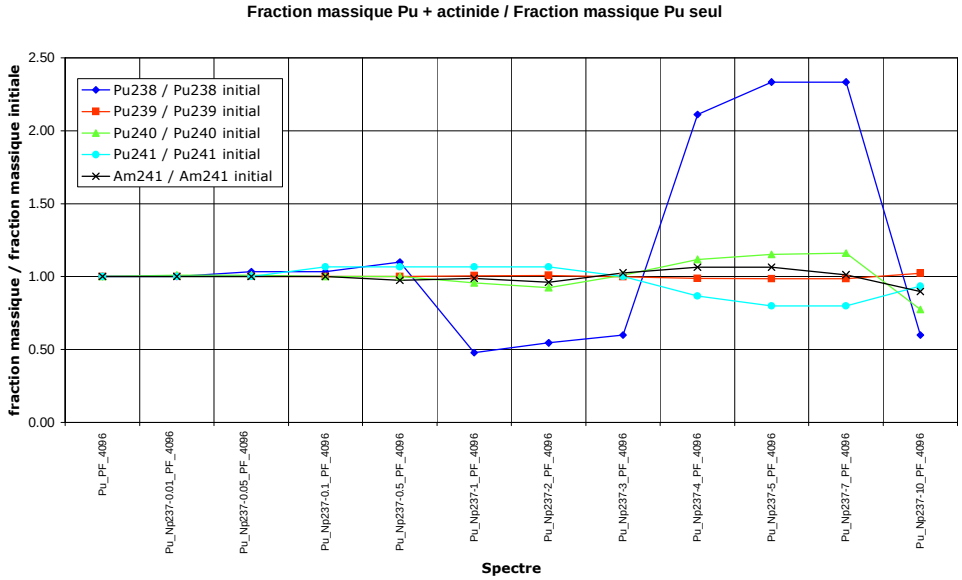

Figure 11: effect of increasing Np-237 contents on the results of IC determination using the IGA software tool. The Pu isotopic composition, measurement conditions, and package configuration are fixed.

8. Conclusion

A tool and procedures to generate complex gamma test spectra from individually measured elementary spectra have now been developed and validated. The system needs some additional work to become fully integrated: specifically, the software which allows operating the bench and processing the measurements needs to be developed on the basis of the procedure described above, and accounting for the lessons learned in the laboratories during this first exercise. This powerful tool could be used to generate test spectra and study a large number of parameters and configurations that could have an impact on waste characterization based on gamma spectrometry. These data would be based on actual experimental data in an energy range where numerical simulation is not easy or very reliable. The system is cheaper, simpler, more reproducible than experiments performed on actual packages or mock-ups, and allows testing an increased number of parameters, with the following advantages:

- Elimination of the package preparation step, and of the associated logistics to combine the packages and the sources and transport them to the measurement stations.
- Decrease of spatial requirements, thus allowing the implementation of three detectors with their electronics in a single room.
- Optimization of the counting time, since the distance between the detector and the source would not be constrained in the same way.
- Improved reproducibility
- Enlarged field of applicability (not limited to the geometry and configuration of the mock-up packages or to the available isotopic mixtures).

For the original objective of the study, which was to determine the field of application of the various software tools for Isotopic Composition in the context of waste management, a preliminary campaign, based on a reduced data base, has now been started. The preliminary results confirm that the methodology provides a means of testing the IC software tools as expected, by identifying levels of interfering radionuclides liable to induce discrepancies. At the end of this exercise, it is expected that the comprehensive results will allow guidelines to be set, help the users perform reliable gamma spectrometry-based Isotopic Composition determinations on waste packages.

A Suggested Improved Wall Thickness Correction for Canberra's IMCA

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Abstract:

For homogeneous bulk compounds of uranium the enrichment may be determined by a direct measurement of the emerging 186 keV gamma-line strength using a spectrometer calibrated against reference material measured in the same geometry. In practice correction factors for the attenuation suffered in the container wall, which may differ in thickness, density and material, to the attenuator used during the calibration must be applied. In the IMCA implementation of the method, this correction is estimated according to Beer's exponential law of removal of the primary photons assuming a planar geometry with normal incidence. In practice the container wall may have a slight curvature over the field of view but more importantly because a close geometry is typically used, rays with a fairly broad range of slant angles are accepted incident on the detector. If the attenuator used in the calibration is a close match to the wall of the unknown measurement items, then one might expect this simplification to be of second order importance; but the meaning of close needs to be quantified. When the differences are larger we need a means of making a more refined correction.

In this paper we address the problem by performing numerical simulations using the ISOCS code for a range of container wall materials and thicknesses with a common collimated Ge-spectrometer arrangement. The result is a revised correction factor which we have tested experimentally.

Keywords: IMCA, enrichment meter, uranium, attenuation correction

1. Introduction

Recently we have reviewed the measurement uncertainties [1], including the treatment of the calibration data [2], associated with the application of the long established enrichment meter principle for the determination of the enrichment of U-235 in thick homogeneous U-compounds. This work is a continuation of our efforts to improve the state of the practice. The determination of enrichment for homogeneous chemical compounds which may be considered infinitely thick in relation to the 1/e removal length for the 186 keV gamma-ray emitted by U-235 is quite simple in principle. This is because when measured with a detector which has a carefully controlled field of view of the item the count rate in the full energy peak is directly proportional to the fractional abundance of U-235 in the compound. The enrichment can then be obtained by comparison of the observed rate to that observed from a reference item of known enrichment. If the attenuation of the container wall is not identical to that of the attenuator used during the calibration stage a correction for the difference must be made. Our objective here is to quantify this correction over a representative range of conditions for a typical field measurement system.

2. Definition of the problem

Canberra's IMCA instrument [3] is used to measure the enrichment of material in large containers. Frequently, the container wall thickness of the measurement item is not exactly the same as the container wall thickness of the standard that is used for calibration. As a result, a correction factor

must be applied. In the IMCA instrument, this correction is estimated according to Beer's exponential law of removal of the primary photons assuming a planar geometry with normal incidence. However, due to the close counting geometry and finite collimator dimensions, the assumption that photons pass perpendicularly through any attenuating materials between the source and the detector (namely the container walls) is not strictly true. Photons pass through the attenuating material over a range of slant angles determined by the collimator's dimensions.

The question to be resolved is how does the actual attenuation that occurs due to photons passing through the attenuator at a variety of slant angles compare with the assumed attenuation, calculated according to Beer's law. In this paper we will evaluate how the attenuation due to photons passing through an attenuator of actual thickness, *t,* over a range of angles as determined by the dimensions of a collimator, compares to the attenuation predicted by Beer's law (i.e., $e^{-\mu t}$, where μ is the linear attenuation coefficient).

We will further show that over the range of conditions typically encountered (wall material, wall thickness, and photon energy) the actual attenuation can be approximated very well by Beer's law by substituting an 'effective thickness' for the actual thickness, and that this effective thickness is simply a constant scaling factor, *s*, times the actual thickness:

$$t_e = s \cdot t \tag{1}$$

where *s* is determined by the dimensions of the collimator, but is virtually independent of the wall material, wall thickness and photon energies for the range of values typically encountered by the IMCA application.

3. Analytical Analysis – Point Detector

For pedagogic purposes, let us begin our analysis by considering the case of a point detector viewing a wide (wider than the collimator's field of view), 'infinitely thick' sample of U_3O_8 through an attenuator of thickness *t* and a cylindrical collimator of length L and aperture radius R situated a distance G from the active face of the detector as shown in Figure 1.

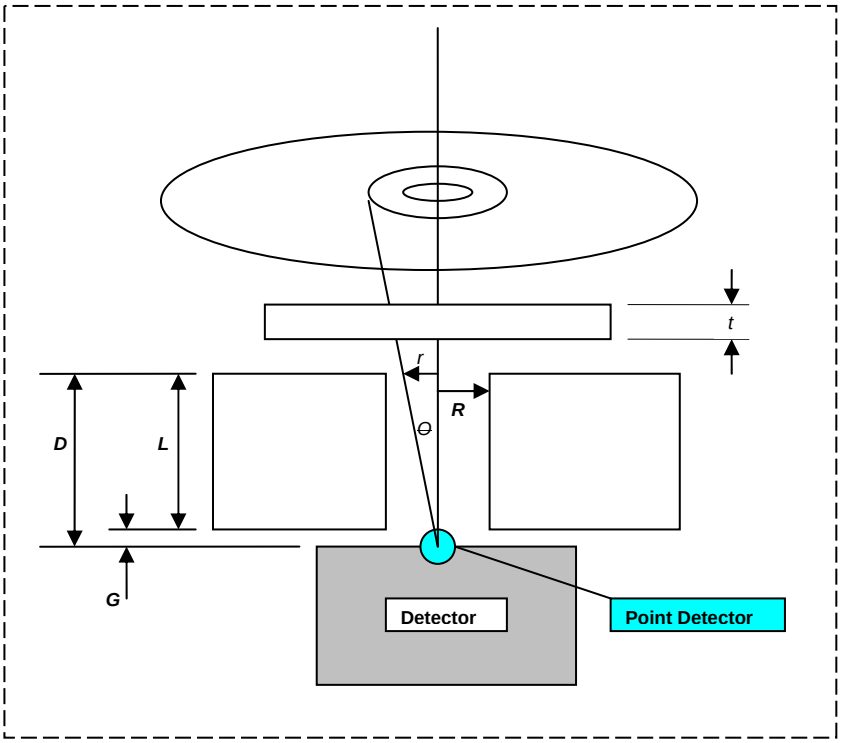

Figure 1: Schematic of the described measurement geometry with critical parameters labeled.

When considering only full energy photons, the assumption of infinite thickness implies that the source strength in every outward direction at every point on the surface of the source is the same (see Figure 2). In other words, it is not necessary to consider the amount of material (both as a source and an attenuator) along the path of a ray through the source as any ray emanating from the surface of the source is due to and attenuated by the same amount of material (linear along the ray path) as shown in Figure 2.

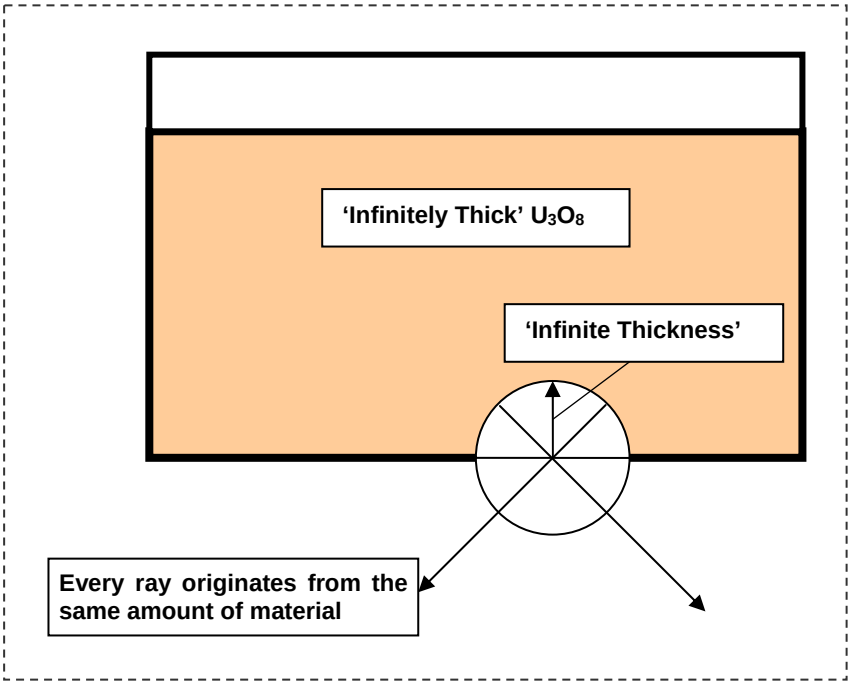

Figure 2: Illustration that all rays 'see' an infinite thickness.

Thus, the source can be treated as a disk with constant source strength, *K*, at all points on the surface, and the weighted average attenuation, <*f*>, is calculated as

$$< f >= \frac{K \cdot \int_{r=0}^{r=R} 2\pi \cdot r \cdot e^{-\mu t/\cos\theta} dr}{K \cdot \int_{r=0}^{r=R} 2\pi \cdot r dr} = \frac{\int_{r=0}^{r=R} 2\pi \cdot r \cdot e^{-\mu t/\cos\theta} dr}{\int_{r=0}^{r=R} 2\pi \cdot r dr} \quad (2)$$

in which the constant source strength in the numerator and denominator cancels as shown. In this expression:

$\theta =$ the slant angle = $\tan^{-1}[\frac{r}{D}]$
$\mu =$ linear attenuation coefficient
$t =$ the physical thickness of the attenuating slab
$r =$ the radial distance from the centerline to the ray measured at the front of the collimator

Noting that

$$\cos(\theta) = \frac{D}{\sqrt{D^2 + r^2}} \quad (3)$$

equation (2) becomes

$$< f >= \frac{\int_{r=0}^{r=R} 2\pi \cdot e^{-\frac{\mu t}{D}\cdot\sqrt{D^2+r^2}} \cdot r \cdot dr}{\int_{r=0}^{r=R} 2\pi \cdot r dr} \tag{4}$$

Equation (4) can be solved using standard forms by defining a new variable, *x*, as follows

$$x = \sqrt{D^2 + r^2} \tag{5}$$

such that

$$dx = \frac{1}{2} \cdot \left[D^2 + r^2\right]^{-1/2} \cdot 2r \cdot dr = \frac{r}{\sqrt{D^2 + r^2}} dr \tag{6}$$

and multiplying the integrand by

$$1 = \frac{\sqrt{D^2 + r^2}}{\sqrt{D^2 + r^2}}$$

so that equation (4) becomes

$$< f >= \frac{2\pi \int_{x=\sqrt{D^2+0^2}}^{x=\sqrt{D^2+R^2}} \cdot e^{a\cdot x} \cdot x \cdot dx}{\pi R^2} \tag{7}$$

where

$$a = -\frac{\mu \cdot t}{D} \tag{8}$$

which has the solution

$$< f >= \frac{2\pi \frac{e^{ax}}{a^2} \cdot (ax - 1)\Big|_{x=\sqrt{D^2+0^2}}^{x=\sqrt{D^2+R^2}}}{\pi R^2} \tag{9}$$

From this weighted average attenuation, we define an ‘effective thickness’, t_e, defined as

$$t_e = -\frac{ln\left[< f >\right]}{\mu} \tag{10}$$

so that

$$< f >= e^{-\mu \cdot t_e} \tag{11}$$

Assuming, for illustrative purposes, aluminum to be the attenuator material and an IMCA collimator (*R*=0.4953 cm; *D*=2.7066 cm), the above equations yield the results shown in Table 1 for the primary lines of U-235 (i.e., 143, 165, 185, 205 keV). Canberra’s ISOCS program, which uses a polynomial

representation of the mass attenuation coefficients of the various elements [4, 5], was used to calculate the linear attenuation coefficients as μ = 0.3642, 0.3475, 0.3319, 0.3203 cm^{-1} for the 4 lines, respectively.

Thickness, cm	RATIO = Scale Factor			
	@143 keV	@163 keV	@185 keV	@205 keV
0.1	1.00832542	1.00832544	1.00832545	1.00832547
1	1.00832165	1.00832184	1.00832202	1.00832216
2	1.00831747	1.00831785	1.00831821	1.00831848
3	1.00831328	1.00831386	1.00831440	1.00831480
5	1.00830491	1.00830588	1.00830677	1.00830744
10	1.00828399	1.00828591	1.00828771	1.00828904
20	1.00824214	1.00824600	1.00824958	1.00825224
30	1.00820031	1.00820609	1.00821146	1.00821545
40	1.00815849	1.00816620	1.00817336	1.00817868
50	1.00811671	1.00812633	1.00813528	1.00814192

Table 1: Scale Factors as derived for the point detector case.

As can be seen from Table 1, the ratio of the effective thickness to the actual thickness (i.e., the Scale Factor), is reasonably insensitive to attenuator thickness or photon energy over the entire range of values typically encountered in the IMCA application. Similar calculations for iron and tin produce similar results; i.e., scale factors in the range of 1.00817 to 1.00832 for thicknesses up to 3 cm. This virtual invariance of the scale factor allows one to define a single value to the scale factor to be used over this range of energies and attenuator materials and thicknesses. While the scale factor is reasonably insensitive to attenuator thickness or photon energy, it does exhibit a slight dependence upon both energy and attenuator thickness. The scale factor increases slightly with increasing energy and decreases slightly with increasing attenuator thickness. These effects can be explained in terms of the characteristics of a weighted average and the concavity of the function being averaged.

For a monotonically decreasing function with positive concavity, *f(x)* (such as the exponential $e^{-\mu \cdot x}$), the effective thickness, t_e, defined as follows:

$$f(t_e) = < f > = \frac{\int_{x_1}^{x_2} x \cdot f(x) dx}{\int_{x_1}^{x_2} x dx} \tag{12}$$

decreases as the **relative** concavity of *f(x)* increases. The relative concavity is defined as

$$f''_{rel} = \frac{f''(x)}{f(x)} \tag{13}$$

4. Analytical Analysis – Extended Detector

It can be inferred that extending the analysis of the point detector to a detector of finite dimensions will yield a larger scale factor because the weighted average will include rays that have longer path lengths through the absorber (i.e., larger slant angles) as can be seen in Figure 3.

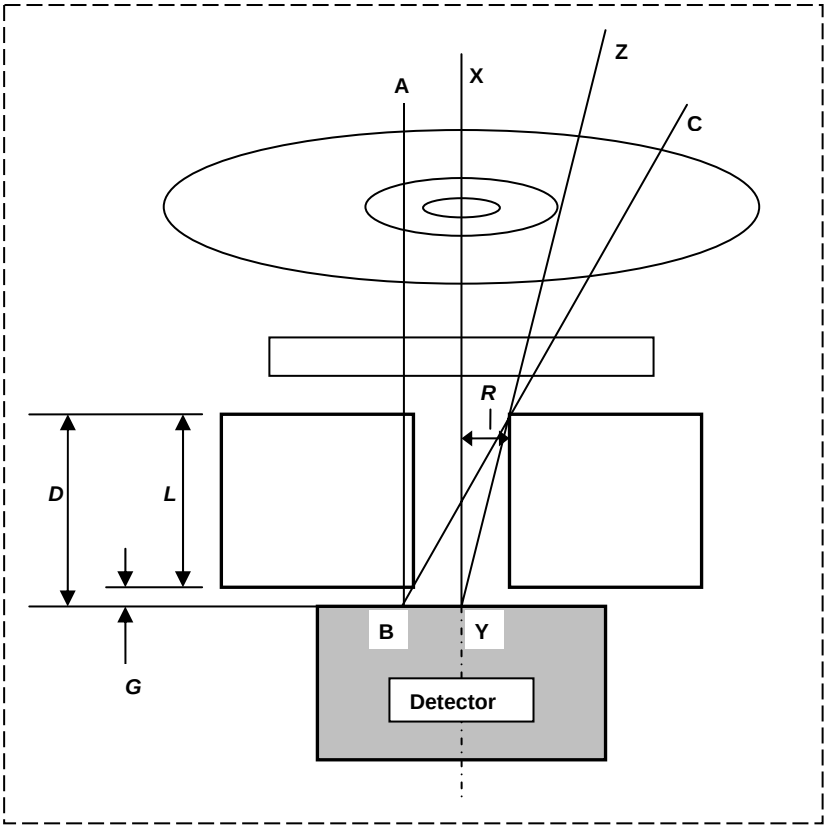

Figure 3: The maximum slant angle for a point detector centered in the aperture of the collimator is XYZ while the maximum slant angle for a detector of finite size is ABC.

For a point detector, the range of slant angles included in the weighted average is from zero to angle XYZ. As one integrates over the face of a detector of finite dimensions, the range of slant angles included in the weighted average is from zero to angle XYZ for the detector surface element at the center of the detector, and increases as one integrates over elements of the detector face further from the center. The maximum slant angle for detector elements that are furthest from the center of the detector face but still visible to the source is angle ABC:

$$ABC = \tan^{-1}\left[\frac{2R}{L}\right] \tag{14}$$

As a result of these longer path lengths being included in the weighted average attenuation, the weighted average will be lower (i.e., greater attenuation). This lower weighted average produces a larger effective thickness, and thus a larger scale factor.

From equation (1), it can be seen that the scale factor associated with a ray passing through an attenuator at an angle of ABC is 1/cos(ABC). This (single ray) scale factor (1/cos(ABC)) is the maximum value that is included in the weighted average, and as such represents an upper limit on the scale factor (s_{Max}). Thus the scale factor for a real detector and collimator of finite dimensions should lie between the scale factor one obtains for a point detector (s_{Point}) and the maximum scale factor, s_{Max}. For the IMCA collimator these values are:

$$s_{Max} = \frac{1}{\cos\left(27.47^0\right)} = 1.127 \tag{15}$$

$$s_{Point} \cong 1.00829 \tag{16}$$

where for s_{Point} we have taken the simple average of the scale factors from Table 1 for energies of 143, 163, 185, and 205 keV and thicknesses $0.1 \leq t \leq 3.0$ cm.

Rather than attempt to obtain a closed form solution for the case of a detector and collimator of finite dimensions, a numerical simulation was used to obtain the scale factors for attenuators of various materials and thicknesses.

5. ISOCS Simulations

Canberra's ISOCS program [5] was used to model a GC1018 detector[1] with an IMCA collimator viewing a NBS SRM 969 (Uranium Isotopic Standard Reference Material) source [6]. The model includes the finite attenuation through the corners of the collimator and the angular dependence of the detector's response. In these important respects it is more faithful then the earlier analytical approximation. However, ISOCS is based on a ray-tracing and so does not account for the small angle scattering effect.

Photopeak efficiencies for this detector/collimator/source configuration were calculated for collimator-face to source distances from zero to 3.0 cm at characteristic energies from U-235 of 143, 163, 185, and 205 keV with and without attenuators. The ISOCS model with a 3 cm attenuator is shown in Figure 4. The collimator assembly shown in the picture includes a tantalum collimator set in a sintered tungsten shield with tin filters.

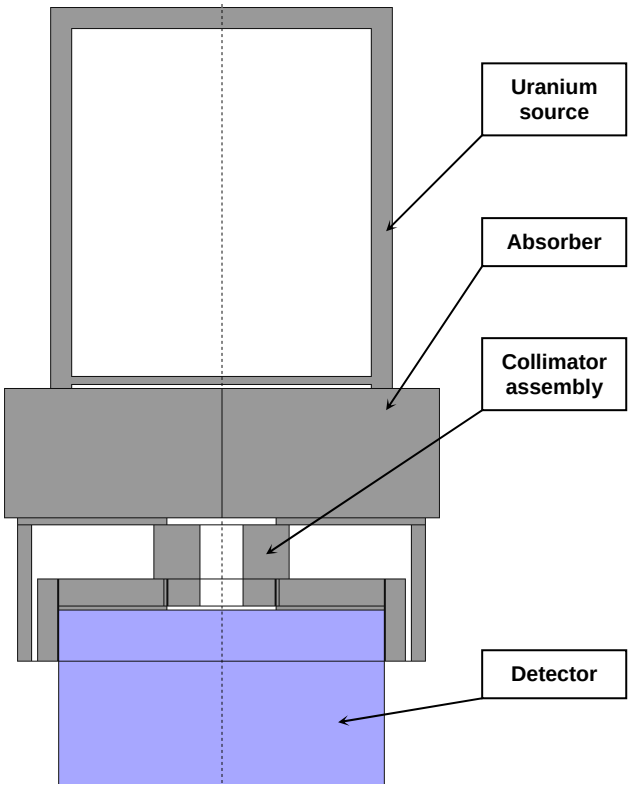

Figure 4: Counting geometry (as modeled and viewed in ISOCS).

The efficiency for counting an infinitely thick wide source through a collimator that provides a well defined narrow view of the source should be practically invariant to distance between the source and the collimator because the area of the field of view increases as distance squared while the efficiency for each point on the surface being viewed decreases (almost) as distance squared. This invariance

[1] The IMCA product supports 3 detector types: NaI(Tl), CdTe, or a LEGe [planar] detector. A model GC1018 Ge coax detector was chosen for the modeling because a characterization file [required for ISOCS] was not available for the type of LEGe normally used with an IMCA. For the present considerations this is not of consequence.

of the efficiency with distance is not strictly valid for a flat source as the path lengths from the detector to the outer ring of the field of view increase with distance as 1/cos(slant-angle). This effect is minimized by designing the collimator to provide a narrow field of view.

Photopeak efficiencies were calculated with ISOCS for collimator-face to source distances of zero to 3.0 cm with and without attenuating material between the ISOCS detector/collimator and the NBS SRM 969 source at 143, 163, 185, and 205 keV. The attenuation due to the absorbers is given by the ratio of the efficiency with the absorber in place to the efficiency without the absorber:

$$Attenuation(E,t) = \frac{\varepsilon_{Al\_absorber}(E,t)}{\varepsilon_{NO\_absorber}(E,t)} \tag{17}$$

This attenuation is essentially the weighted average of the attenuation for each ray that is initially directed such that it would pass through the collimator. From these attenuation values an effective thickness, t_e, is calculated according to equation (18):

$$t_e(E,t) = -\frac{\ln[Attenuation(E,t)]}{\mu} \tag{18}$$

As was done for the case of the point detector analysis, the scale factor, *s*, is defined as the ratio of the effective thickness to the actual thickness, t_{act}:

$$s(E,t) = \frac{t_e(E,t)}{t_{act}} \tag{19}$$

Scale factors for aluminum absorbers of thicknesses from zero to 3.0 cm at energies of 143, 163, 185, and 205 keV are presented in Table 2. Similar calculations for iron and tin absorbers produce the scale factors shown in Tables 3 and 4, respectively.

Thickness, cm	**Scale Factor**			
	@143 keV	**@163 keV**	**@185 keV**	**@205 keV**
0.00	n/a	n/a	n/a	n/a
0.10	1.0331	1.0341	1.0355	1.0372
0.20	1.0329	1.0336	1.0347	1.0363
0.30	1.0319	1.0327	1.0340	1.0357
0.40	1.0314	1.0322	1.0334	1.0351
0.50	1.0311	1.0318	1.0329	1.0348
1.00	1.0300	1.0312	1.0328	1.0347
1.50	1.0321	1.0330	1.0344	1.0360
2.00	1.0322	1.0330	1.0346	1.0361
2.50	1.0319	1.0329	1.0343	1.0358
3.00	1.0324	1.0332	1.0344	1.0357
Average =	**1.0319**	**1.0328**	**1.0341**	**1.0358**
Range about average [+] =	0.11%	0.13%	0.13%	0.14%
Range about average [-] =	-0.18%	-0.15%	-0.13%	-0.11%
Midpoint (MP)=	**1.0315**	**1.0326**	**1.0341**	**1.0359**
Range about MP = +/-	0.15%	0.14%	0.13%	0.12%

Table 2: Scale Factors as determined using ISOCS for Al absorbers and a real detector.

Thickness, cm	Scale Factor			
	@143 keV	@163 keV	@185 keV	@205 keV
0.00	n/a	n/a	n/a	n/a
0.10	1.0334	1.0343	1.0358	1.0375
0.20	1.0332	1.0339	1.0350	1.0366
0.30	1.0321	1.0329	1.0342	1.0359
0.40	1.0316	1.0323	1.0336	1.0353
0.50	1.0312	1.0320	1.0331	1.0350
1.00	1.0300	1.0312	1.0328	1.0347
1.50	1.0318	1.0328	1.0342	1.0358
2.00	1.0318	1.0326	1.0342	1.0358
2.50	1.0313	1.0324	1.0338	1.0353
3.00	1.0315	1.0325	1.0337	1.0351
Average =	**1.0318**	**1.0327**	**1.0340**	**1.0357**
Range about average [+] =	0.16%	0.16%	0.17%	0.17%
Range about average [-] =	-0.18%	-0.15%	-0.12%	-0.10%
Midpoint (MP)=	**1.0317**	**1.0328**	**1.0343**	**1.0361**
Range about MP = +/-	0.17%	0.15%	0.14%	0.14%

Table 3: Scale Factors as determined using ISOCS for Fe absorbers and a real detector.

Thickness, cm	Scale Factor			
	@143 keV	@163 keV	@185 keV	@205 keV
0.00	n/a	n/a	n/a	n/a
0.10	1.0334	1.0343	1.0358	1.0375
0.20	1.0330	1.0338	1.0350	1.0365
0.30	1.0319	1.0327	1.0341	1.0359
0.40	1.0312	1.0321	1.0334	1.0352
0.50	1.0307	1.0317	1.0329	1.0348
1.00	1.0291	1.0305	1.0323	1.0343
1.50	1.0303	1.0317	1.0334	1.0352
2.00	1.0299	1.0312	1.0332	1.0349
2.50	1.0290	1.0307	1.0326	1.0345
3.00	1.0287	1.0303	1.0322	1.0339
Average =	**1.0307**	**1.0319**	**1.0335**	**1.0353**
Range about average [+] =	0.26%	0.24%	0.22%	0.22%
Range about average [-] =	-0.20%	-0.15%	-0.12%	-0.13%
Midpoint (MP)=	**1.0310**	**1.0323**	**1.0340**	**1.0357**
Range about MP = +/-	0.23%	0.19%	0.17%	0.18%

Table 4: Scale Factors as determined using ISOCS for Sn absorbers and a real detector.

As can be seen from Tables 2, 3 and 4, the scale factor is reasonably constant for the range of absorber materials, thicknesses and energies anticipated for the IMCA application, as was previously observed for the point detector case. It is thus concluded that one can represent the attenuation due to photons passing through an attenuator with a range of slant angles - as occurs when one views a source through a collimator – by Beer's law by substituting an 'effective thickness' for the actuall thickness, and that this effective thickness is simply a constant scaling factor, *s*, times the actual thickness:

$$t_e = s \cdot t \tag{20}$$

Furthermore, for the dimensions of the IMCA collimator, the value of *s* over the full range of materials (Al, Fe and Sn) and energies considered (143 keV to 205 keV) is 1.0333[2]. For 185.7 keV specifically, the value of s is 1.0339[3].

6. New Correction Factor

From the physical arguments and calculations presented above we are led to conclude that the use of the simple attenuation model currently embodied in IMCA is adequate, but an effective thickness ($t_e = s \cdot t$) or, equivalently, an effective linear attenuation coefficient ($\mu_e = s \cdot \mu$) should be used. Since *s* depends on the geometry and book values of μ vary from evaluation to evaluation, a pragmatic approach is to determine μ_e values specific to the implementation.

7. Experimental Tests

A set of experimental measurement has been performed in order to experimentally determine the scale factor, *s*. Two types of measurements were done using a small LEGe detector (see Figure 5 below).

In the first experiment the “parallel beam” geometry was set up using two 2.5 cm thick lead blocks with 0.5 cm diameter holes. One block was set directly in front of the detector and the other at 10 cm away with holes aligned. A “parallel beam” transmission, T_{paral}, of the 186 keV gamma-rays after passing through three different absorbers was measured using a small 1g uranium source (~97% enrichment). A 1.8cm thick aluminum, 0.9cm thick steel and 0.3175cm thick nickel plates were used individually as absorbing materials. Due to the low intensity of the source, the counting time varied from about 19 to 70 hrs for different absorbers.

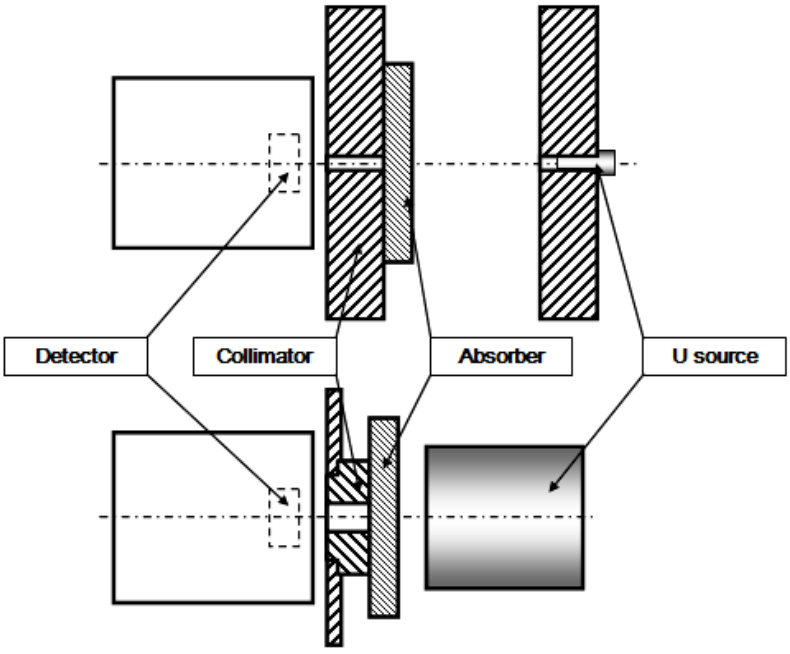

Figure 5: Geometries used for experimental determination of the scale factor.

[2] This value of *s* is the average of all the values of *s* in Tables 2, 3 and 4.
[3] This value of *s* is the average of all the values of *s* for 185 keV from Tables 2, 3 and 4.

In the second set of measurements a standard IMCA collimator was used. This collimator is made of tantalum and has an aperture of about 1cm in diameter. A 4.46% enriched uranium standard (NBS SRM969) was used to measure a gamma-ray transmission for a collimated geometry, T_{coll}. The counting time varied from 2 to 17 hours for different absorbers.

The gamma-ray transmission for the two measured geometries (T_{paral} and T_{coll}) can be represented by equations (21) and (22) given below.

$$ln\, T_{paral} = -\mu \cdot \rho \cdot t \tag{21}$$

$$ln\, T_{coll} = -\mu \cdot \rho \cdot s \cdot t \tag{22}$$

where:

μ = mass attenuation coefficient, cm^2/g
ρ = absorber density, g/cm^3
t = absorber thickness, cm
s = scale factor

As a result, the scale factor can be simply determined using equation (23). In this approach, it is not necessary to know the exact thickness of an attenuating material nor the book values for the attenuation coefficient since the quantities cancel out.

$$s = \frac{ln\, T_{coll}}{ln\, T_{paral}} \tag{23}$$

In each case, the transmission, T, can be directly measured by taking a ratio of the count rates in the 186 keV peak obtained with and without an absorber (CR_{abs}^{186} and $CR_{no\ att}^{186}$) with minor live time corrections.

$$T = \frac{CR_{abs}^{186}}{CR_{no\ att}^{186}} \tag{24}$$

The experimentally determined scale factor for three different absorbing materials is shown in Table 5.

Absorber	Scale Factor	
1.8 cm aluminum	1.032	+/- 0.022
0.9 cm steel	1.020	+/- 0.012
0.3175 cm nickel	1.028	+/- 0.029
Weighted average	**1.023**	**+/- 0.010**

Table 5: Scale Factors as determined using experimental measurements with various absorbers.

Most of the uncertainty in the results comes from the statistical uncertainty in the 186 keV peak area and could be significantly improved if a highly enriched uranium standard was used in place of the 4.46% enriched item. In principle, counting time can also be extended and attenuator thickness optimized.

8. Discussion

Our closed form solution for the point detector case suggests that one should be able to define a single scale factor that applies to the range of energies, absorber thicknesses, and materials normally encountered in the IMCA application. Our numerical simulation produces results that are consistent with the point detector case – namely that the scale factor remained reasonably constant over the range of energies, absorber thicknesses, and materials modeled. Our experimental results agree with our numerical simulations, although the experimental results are less precise as we would hope. In principle this is not a limitation.

The value one derives for the scale factor is obviously very sensitive to the value assumed for the mass attenuation coefficient (and the density) of the material. When comparing mass attenuation coefficients from sources such as XCOM to MCNP library values or to the values generated by the ISOCS polynomial representation, one finds that they agree to within about 1%. However, one percent uncertainty in what amounts to a three percent effect (a scale factor of 1.03 implies that the effective thickness or effective attenuation coefficient is 3% greater than what one would obtain from a parallel beam case) is very significant. For these reasons it was thought that the best method of determining the attenuation coefficient for a test material would be to measure it using a far field parallel beam geometry.

Our objective was to extract a scale factor, *s*. However in practical application what is needed is μ, and this can be determined far more easily from a transmission vs thickness experiment in the actual geometry.

9. Conclusion

The use of the simple wall attenuation correction factor within IMCA has been justified provided it is applied with effective quantities (thickness or linear attenuation coefficient). We have calculated the appropriate scaling factor for one geometry (collimator). An experimental approach to finding the scaling factor directly has also been explained.

We note that when the calibration is performed with a wall similar to that of the unknown item, the uncertainty in the correction factor is kept small. Likewise the effective linear attenuation coefficient can be readily inferred from calibrations made with two or more wall thicknesses, so that instrument specific interpolation can be accurately performed.

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Development of list mode neutron coincidence data acquisition system

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Abstract:

The list mode acquisition is a relatively new way of neutron coincidence counting. It is based on the recording of the follow-up times of neutron pulses from the neutron counter into a file on the hard disk. The recorded pulse train can be analysed with dedicated software during the measurement or later.

Hardware and software for list mode neutron coincidence acquisition have developed in the Institute of Isotopes. The external hardware is based on common FPGA circuits and transfers time-intervals between consecutive neutron pulses to the PC via USB line. Follow-up data are written in binary format to hard disk by a PC program. During measurement the time interval distribution is displayed, what is useful for diagnostics and research.

Software is developed for calculating multiplicity distributions as well as total and coincidence count rates. The evaluation is possible with different pre-delay, gate length and long delay, and the evaluation software is very fast. Also the Rossi-alpha distribution and a corrected die away time can be calculated.

The performance of the hardware and the software was investigated and test measurements were carried out using a conventional multiplicity shift register (JSR-14) and recording the pulse trains by a list-mode card (IKI-card) in parallel.

Keywords: NDA, neutron coincidence counting, list mode

1. Introduction

List mode acquisition is a relatively new way of neutron coincidence counting. In opposite to multiplicity counting data acquisition and evaluation are separated. A time stamp of each neutron is recorded and saved into a file. By this method the same pulse train can be evaluated with different programs.

List mode systems usually consist of a PC. A stand alone data acquisition version is also possible, but it would be somewhat impractical because of the huge data files. Systems built with a PC should consist of three main parts:

- A hardware unit making time stamps and with sufficient buffer data space and I/O capability for high speed data transfer
- Data acquisition software with suitable user interface for reading in and saving data to hard disk
- Evaluation software for calculating Singles, Doubles and Triples coincidence values and displaying follow-up time and Rossi-alpha distributions.

2. Hardware

A list mode system can be either an external unit or a PC extension card. For practical purposes we decided to make an external unit. So it can be used even with a laptop.

An external unit must have a data connection to the PC. Regarding a maximal data rate of some million impulses per second this must be a USB 2.0 data line.

2.1 Synchronization

Incoming detector impulses are asynchronous to clock pulses, so they must be synchronized. Synchronization distorts measured follow-up times. The error is approximately one half clock period. This has a minor effect on calculated coincidence values. At high clock rates this effect is negligible.

Another problem is when there are more neutron impulses in a clock period. Traditional solution of this problem is a so called de-randomizer circuit. It is essentially a fast buffer, which holds incoming impulses until the slower processing circuitry can handle them. A modern FPGA has sufficient high processing rate, so there is no need for a dedicated de-randomizing circuit.

2.2 Time stamping

Working with usual real number time stamps requires much hardware and software resources. We realised this at the ESARDA neutron coincidence evaluation benchmark test and use since then follow-up times. That means the absolute value of time stamps is replaced by a relative value measuring the time interval between consecutive impulses. Follow-up times can be represented by integer numbers. This is beneficial both in required storage space and processing time.

Follow-up times are simply represented by the number of clock pulses passed between two consecutive neutron impulses. Of course the data file must contain the cycle time. This is in our case 10 ns which corresponds to a clock frequency of 100 MHz.

2.3 Buffer

Incoming neutron impulses have a statistical fluctuation but data transfer takes place in more or less uniform distributed data packets. Besides this there is an even more concerning fact. USB specification does not guarantee error correction and delivery rate at the same time. That means not only the effect of statistical fluctuation must be overcame but also some ‘drop outs’ caused by USB and Windows.

For this reason a large buffer with appropriate control logic is needed. The control should ensure that buffered data are transferred as soon as possible.

In order to avoid buffer overflow at high count rates, data are compressed before buffering. This multiplies effective buffer capacity up to three times at high count rates.

2.4 Implementation

We have implemented our system on a Spartan3 FPGA developing board equipped with one MB buffer memory and a USB 2.0 interface for data transfer. After the one channel instrument a multichannel version was also developed. This performed so good that we use it also for one channel measurements by disabling all but one input channels.

The newest version has a high voltage supply for the JCC-31 detector. The supply is controlled by the data acquisition software.

3. Data acquisition software

The PulseTrainRecorder software reads follow-up values from the PTR-02 hardware through USB line and saves them on hard disk. Besides data saving it offers several practical possibilities, such as:

- Repeated measurements
- Graph of follow-up distribution is expandable and collapsible even while data acquisition
- Reading in and displaying previously recorded binary data files
- Calculating totals and coincidence rates Doubles and Triples
- Calculating Rossi-α distribution and die-away time

Latter two are actually separate programs, which can be used also alone. For the sake of convenience they can be started from within the data acquisition software, too.

4. Evaluation software

Evaluation software is processing list mode data files after measuring. They have to be fast, not to make additional large delay. They have also to offer variable parameters for evaluation.

Coincidence values and Rossi-α distribution are calculated by two separate programs, because latter requires much more time and is not always needed.

Data files consist of a header block and binary data. The header block contains three kinds of data: structure identifiers, auto filled-in fields and user entered description fields. Binary data are consecutive follow-up times in four-byte integer format.

4.1 Coincidence rate calculation

Coincidence rates are calculated in a fractional part of the acquisition time. Processing time never exceeds a few percent of acquisition time even at high count rates.

Total and coincidence rates needed for solving point model equations are calculated as

$$S = \frac{\sum_{i=0}^{n} (R+A)_i}{T_{meas}} = \frac{\sum_{i=0}^{n} (A)_i}{T_{meas}}$$

$$D = \frac{\sum_{i=1}^{n} i(R+A)_i - \sum_{i=1}^{n} i(A)_i}{T_{meas}}$$

$$T = \frac{\sum_{i=2}^{n} \frac{i(i-1)}{2}\left[(R+A)_i - (A)_i\right] - \frac{\sum_{i=1}^{n} i(A)_i}{S \cdot T_{meas}}\left[\sum_{i=1}^{n} i(R+A)_i - \sum_{i=1}^{n} i(A)_i\right]}{T_{meas}}$$

where S, D and T denotes Singles, Doubles and Triples rate respectively. R+A and A are measured multiplicity values for the two windows and T_{meas} is measuring time. Multiplicity distributions of A and R+A windows and calculation results can be saved in a text file.

The Neutron program took part in the Neutron Coincidence Benchmark Test and outraged with its speed. Predelay, gate width and long delay can be set prior to calculation. The same data set can be evaluated with different parameters.

4.2 Rossi-α distribution

The Rossi-α distribution describes the detection probability of another neutron after a trigger event in function of time. Random events have a uniform distribution whereas fission neutrons are time correlated and are usually described by an single exponential term.

Dieaway calculation is made by fitting $N(t) = A + R \cdot e^{-t/\tau}$. The calculated distribution is 1024 μs long with 100 ns time bins. Distribution values and calculation results can be saved in a text file.

5. Comparison

Test measurements were carried out on Pu/Be sources with added smaller californium sources. A commercial JSR-14 multiplicity shift register and PTR-02 card measured the same pulse train by connecting JSR-14 to the copy output of PTR-02. In case of the multichannel version the six preamplifier outputs were used. The other ten inputs of the hardware were shorted out. The table below shows comparison measurement data. Rates are not corrected in order to show the effect of multichannel measurement.

The used detector JCC-31 has a rather low efficiency, therefore only a maximal impulse rate of about 750 kcps could achieved. This was just enough to show, that the net D/S ratio measured by the multichannel version is significantly higher than in the one channel case.

		JSR-14		One channel		Multichannel		Corrected rate
		cps	St.dev.	cps	St.dev.	cps	St.dev.	
Pu/Be (425)	S	327382	37	327382	23	332 284	24	332 276
	D	2013	20	2 083	143	2 076	147	
Pu/Be (425) + Cf-244	S	356059	25	356 610	24	362 612	25	361 873
	D	6770	177	6 608	156	7 153	161	
	D/S	*1,85%*	*0,05%*	*1,85%*	*0,04%*	*1,97%*	*0,04%*	
Pu/Be (701) + Cf-244	S	750518	36	752 682	35	778 732	36	778 850
	D	12641	362	12 859	302	14 688	322	
	D/S	*1,68%*	*0,05%*	*1,71%*	*0,04%*	*1,89%*	*0,04%*	

5. Conclusion

Studying comparison data gives following conclusions

- One channel version is in good agreement with JSR-14.
- At high count rate multichannel version compensates for impulse loss resulting from merging of preamplifier signals

Results show that the PTR-02 device could be a valuable tool for list mode neutron multiplicity measurements.

SESSION 10

CONCEPTS FOR REGIONAL SYSTEMS AND INTEGRATED SAFEGUARDS

Experience in Evaluation of Integrated Safeguards Implementation

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Abstract:

The IAEA Secretariat performs evaluation of safeguards implementation and produces the Safeguards Implementation Report (SIR), which is submitted annually to the Board of Governors. The SIR includes the safeguards statement for the year concerned, in which safeguards conclusions are reported for all States with safeguards agreements in force; it also reports on any case of non-compliance of a State with its undertakings under the relevant safeguards agreement.

For a State with a comprehensive safeguards agreement (CSA) and an additional protocol (AP) in force, and for which a conclusion has been drawn that all nuclear material in the State remains in peaceful use, a State-level integrated safeguards approach (SLA) is developed using three generic State-level objectives and State-specific objectives. An SLA defines the verification activities necessary for meeting the objectives, taking into account the State-specific features, such as the effectiveness of the SSAC and the features of the State nuclear fuel cycle. On the basis of this SLA and taking into account any recommendation for follow-up activities to address safeguards issues, anomalies, questions and inconsistencies identified in the State evaluation process, the Secretariat develops an annual implementation plan (AIP) for the State concerned.

Evaluation of integrated safeguards (IS) implementation is performed on the basis of the SLA and the AIP. The evaluation assesses three generic State-level objectives of IAEA verification activities, common to every State with a comprehensive safeguards agreement:

A. To detect undeclared nuclear material and activities in the State as a whole;
B. To detect undeclared production or processing of nuclear material at declared facilities;
C. To detect diversion of declared nuclear material.

The implementation of IS began in 2001 with Australia; in 2002, IS started in Norway and in 2003, in Indonesia. In 2004, 2005, 2006, 2007 and 2008, respectively 3, 7, 9, 14 and 25 States were under IS for the complete year. In 2008, IS was implemented during the year in eight additional States. For the whole year 2009, IS is being implemented in 36 States. The evaluation is performed on an ongoing basis throughout the calendar year. The results of the IS evaluation are presented, including the savings due to the implementation of IS.

This paper describes the experience and evolution of IS evaluation from 2001 till 2008 and presents an introduction to the future evaluation of IS.

Keywords*: integrated safeguards; safeguards evaluation; safeguards State-evaluation process*.

1. Introduction

The aim of integrated safeguards is to provide the most efficient means to realize the full effectiveness of the strengthened safeguards measures. Integrated safeguards are an optimized combination of all safeguards measures available to the IAEA under comprehensive safeguards agreements and additional protocols, which achieves the maximum effectiveness and efficiency within available resources in exercising the IAEA's right and fulfilling its obligation in paragraph 2 of INFCIRC/153(Corrected).

To achieve this, the IAEA implements measures available under comprehensive safeguards agreements and additional protocols, providing credible assurance of both the non-diversion of declared nuclear material from peaceful nuclear activities and of the absence of undeclared nuclear material and activities in the State as a whole.

Integrated safeguards are not implemented in a State until the initial conclusion of the absence of undeclared nuclear material and activities has been drawn, in addition to the conclusion in respect of non-diversion of declared nuclear material. The IAEA seeks to re-affirm these conclusions annually both as an objective itself and as a condition for the continued implementation of integrated safeguards in that State.

Key to the process by which safeguards conclusions are drawn is the State-level evaluation process, including the preparation of a State evaluation report (SER) and its review by the interdepartmental Information Review Committee (IRC).

For the implementation of integrated safeguards in a State — due to the enhanced assurance in the absence of undeclared nuclear material and activities for the State as a whole — the frequency and intensity of inspection activities at declared nuclear facilities and locations outside facilities (LOFs) may be at a lower level than those defined in the Safeguards Criteria. The Criteria are used to plan and evaluate safeguards in States with no integrated safeguards implemented.

The evaluation of safeguards implementation covers all verification activities performed by the Secretariat in the field and at Headquarters and is performed in order to determine the extent to which the safeguards objectives, outlined below, have been achieved during safeguards implementation for each State in a given year. The evaluation is performed on an ongoing basis throughout the calendar year. The results of the evaluation are reported in the SIR and provide a basis for the Board of Governors' consideration of the safeguards conclusions and assessment of the effectiveness of safeguards implementation.

In order to facilitate the evaluation and comparison of the results for different States, a common set of three generic State-level safeguards objectives, applicable to each State, are defined as follows:

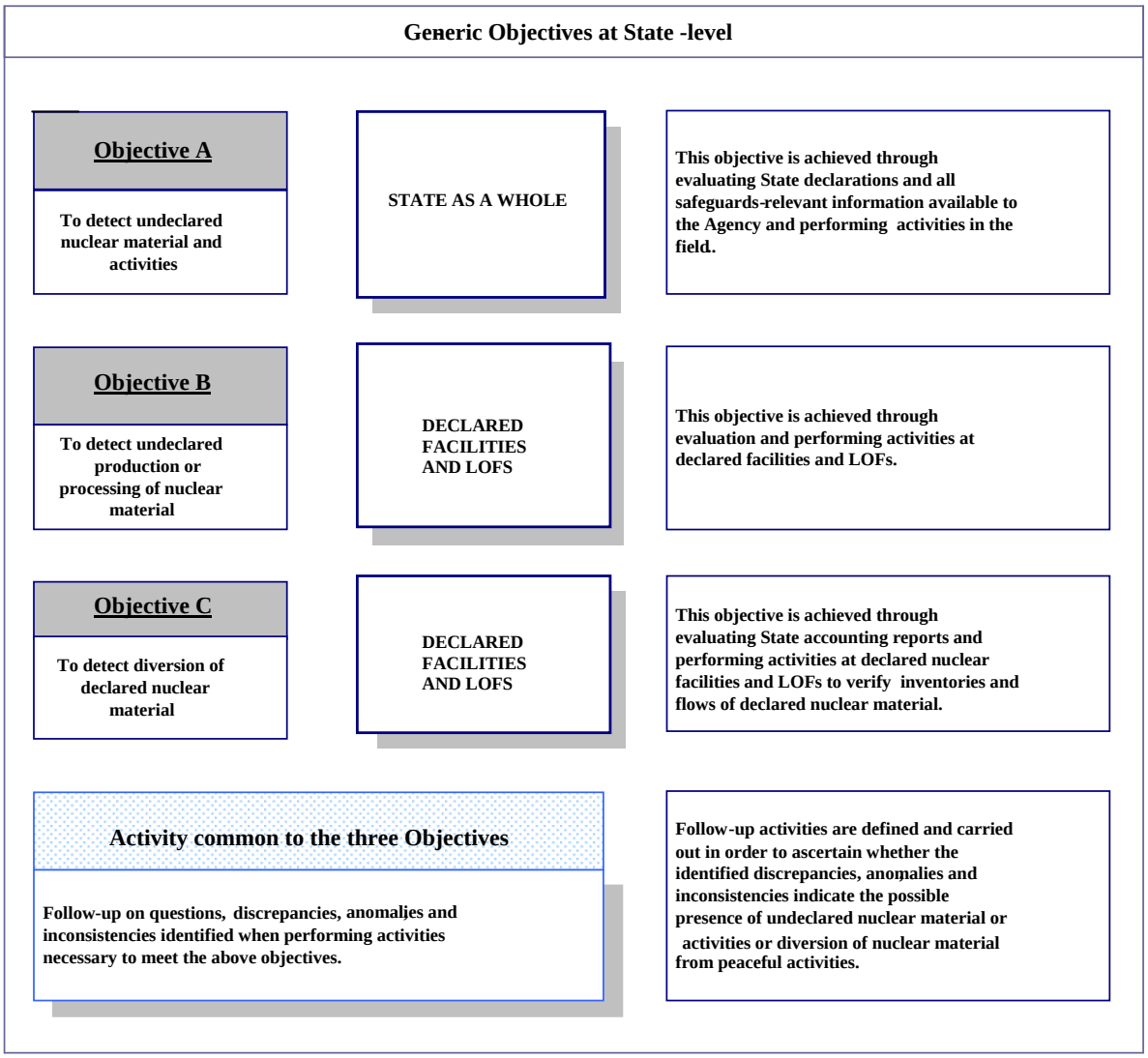
A. To detect undeclared nuclear material and activities in the State as a whole;
B. To detect undeclared production or processing of nuclear material at declared facilities; and
C. To detect diversion of declared nuclear material.

Although the three generic State-level objectives are the same for all States, it is important to establish State-specific objectives for each State. Among these, State-level technical objectives (SLTOs) have to be defined for a specific State based on an analysis of the acquisition paths (i.e. the routes to acquire nuclear weapons usable material), the State-specific nuclear fuel cycle features and the characteristics and other information provided in the safeguards State evaluation report (SER), and to identify safeguards activities accordingly for the State.

2. State-level Concept for the Implementation and Evaluation of Safeguards

2.1 State-level objectives and implementation of safeguards

The IAEA plans and implements the verification activities in the field and at Headquarters to be able to meet the three generic State-level objectives, as described in the figure below.



Although these objectives are interrelated, their separate analysis facilitates the planning and evaluation of safeguards implementation. The follow-up on questions, discrepancies, anomalies and inconsistencies when performing activities necessary to meet the objectives is an activity common to the three objectives.

- Objective A can be fully met only for a State in which the provisions of its additional protocol are implemented. Only for such a State is the IAEA able to draw a broader conclusion that all nuclear material in the State has remained in peaceful activities.
- Objectives B and C are achieved through evaluating all relevant information and, where applicable, implementing inspection activities at declared facilities and LOFs.

The three generic State-level objectives are sub-divided into SLTOs in relation to the acquisition paths and the indicators introduced in the physical model which describes every process and technology capable of producing weapons-usable material (i.e. the acquisition paths for high enriched uranium

(HEU) and separated plutonium) and identifies indicators of the existence or development of a particular process.
The IAEA has defined a list of 42 SLTOs for the detection of undeclared nuclear material and activities and the misuse of declared facilities and LOFs. A 43rd SLTO concerns the detection of diversion of declared nuclear material, which is covered by the traditional safeguards measures under a CSA. The SLTOs are defined by where the objectives have to be addressed and what nuclear material/activity would be involved.

Where integrated safeguards are implemented, all the verification activities in the field or at Headquarters necessary to meet these objectives are defined in the SLA and the AIP. Where no integrated safeguards are implemented, the relevant inspection activities are those defined in the Safeguards Criteria.

2.2 State-level Approach (SLA)

The SLA for a State sets out the safeguards activities to be conducted for a State, in accordance with the conceptual framework for integrated safeguards, comprising both in-field activities and work at Headquarters. It establishes the SLTOs which determine the level and focus of safeguards activities needed for the IAEA to draw soundly based safeguards conclusions. It takes into account features and characteristics of the State's nuclear activities and capabilities identified in the SER, the State-specific acquisition paths, the IAEA's experience in the State, the State-specific conditions for the implementation of safeguards measures (including the use of advanced safeguards technology and the use of unannounced or short notice inspections) and the opportunities for cooperation with the State or regional system of accounting for and control of nuclear material (SSAC or RSAC) in implementing safeguards.

The term 'conceptual framework' is used to describe the set of safeguards concepts, approaches, guidelines and criteria that govern the design, implementation and evaluation of IS.

The State-level approach considers the State as a whole, and assesses wider aspects of a State's nuclear activities, such as:

- the structure of the nuclear fuel cycle, from uranium mines to nuclear waste repositories;
- the nature of fuel cycle-related research and development;
- the manufacture and export of sensitive nuclear-related equipment and material;
- the effectiveness of the SSAC; and
- the optimization of safeguards at facilities, including considerations of grouping of facilities that are related by location, type or function.

The SLA includes a plan for implementing complementary access (CA) at nuclear sites and other locations. The main elements of an SLA for a State are inspection, design information verification (DIV), complementary access, and information collection, review and evaluation.

In summary, the steps to design an SLA are:

- analysis of the State-specific acquisition paths;
- definition of the SLTOs under the three generic objectives A, B and C;
- definition of indicators;
- establishing the safeguards measures and activities to be implemented at HQs and in-field to meet the SLTOs.

The SLA is reviewed by the departmental State-level Integrated Safeguards Committee (SISC) for consistency with approved guidelines and approaches and is approved by the DDG-SG for implementation. The SLA has to be periodically updated on the basis of experience, change in model approaches and technology development.

2.3 Annual Implementation Plan (AIP)

The IAEA develops an AIP for each State on the basis of the SLA. The purpose of the AIP is to identify the actual activities to be performed in a particular calendar year for a specific State, such as:

- inspection activities;
- CA plan;
- DIV activities;
- information analysis at Headquarters; and
- list of questions, inconsistencies or anomalies for follow-up and resolution.

The safeguards activities planned for each year are to be specified in the AIP, which will reflect the SLTOs and associated safeguards measures, both in the field and at Headquarters.

The purpose of the AIP, in addition to identifying the safeguards activities, is to provide a means of converting the non-routine State-level safeguards activities and SER recommendations into scheduled activities.

There are three parts to the AIP:
1. Nuclear material accountancy (NMA) verification and DIV: the facility-based safeguards activities.
2. CA: location, purpose and timeframe for CA activities.
3. Headquarters activities: specific activities outlined in the SLA or recommended in the SER (or/and by the IRC) that are to be performed.

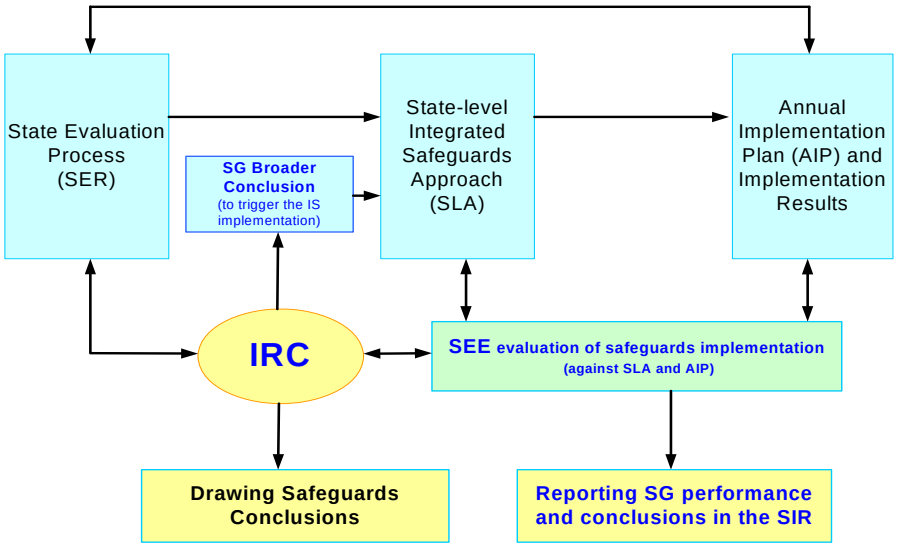
2.4 State-level Evaluation Process

The diagram below shows the interrelations between SIR, SER, SLA and AIP and the State-level evaluation process to enable the IRC to draw conclusions about a State.

The SIR is the IAEA Director General's annual report to the Board of Governors on the work of the Department of Safeguards, including its safeguards conclusions according to States' safeguards undertakings.

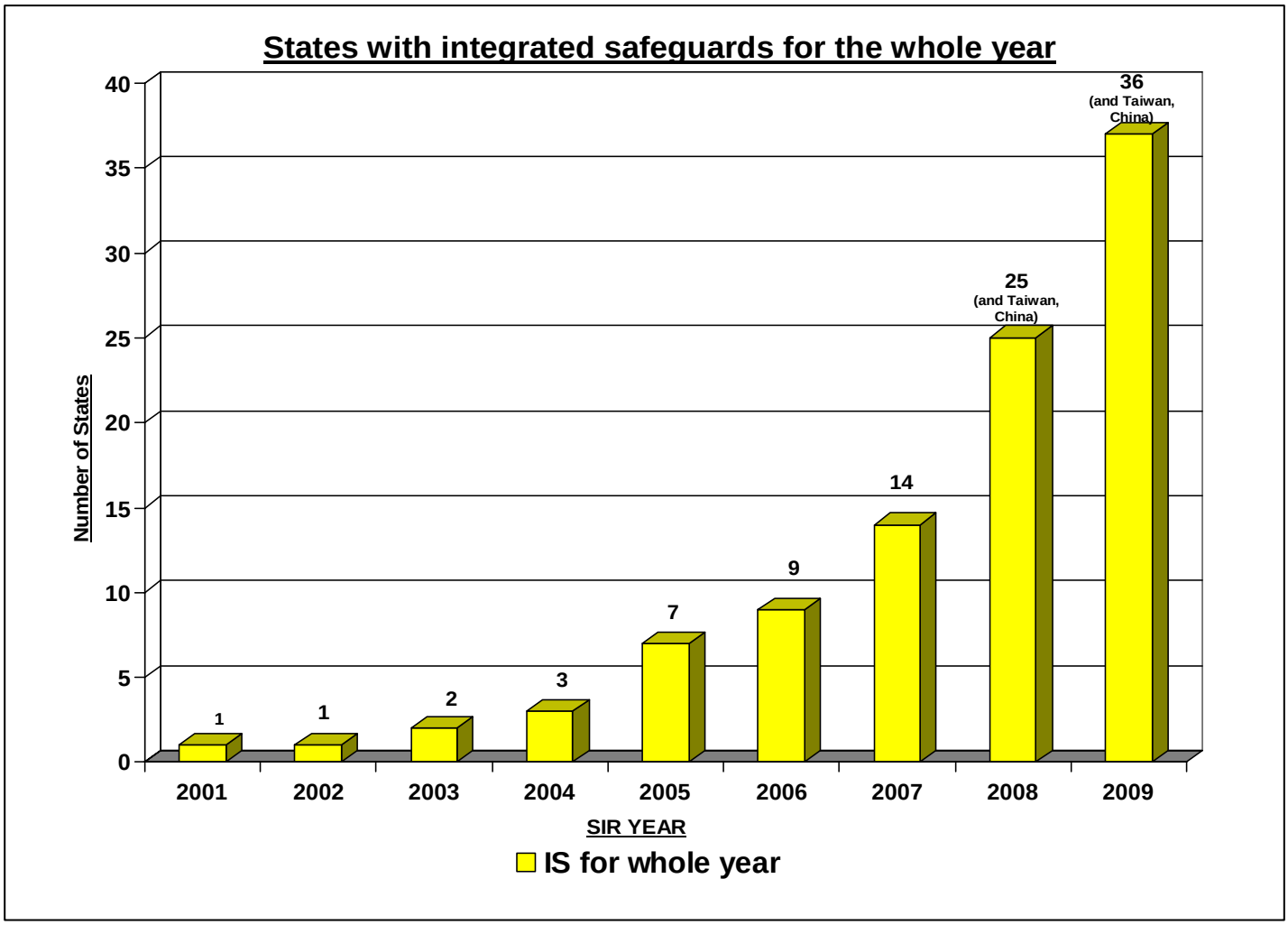
Under a State-level concept, the focus of the safeguards system has shifted from the facility-level to the evaluation of a wider range of information related to the nuclear related programme of each State as a whole. For most States, an SER is produced each year. This report provides a snapshot of the IAEA's knowledge and understanding in respect of the State.

The SER includes an analysis of the State's declarations for internal consistency and for consistency with verification results and all other information available to the IAEA. The SER also contains information on the status of previously identified follow-up actions. When reviewing an SER, the IRC considers the significance of each finding and the extent to which it may affect the basis upon which the safeguards conclusions are drawn and makes recommendations for future follow-up actions accordingly.

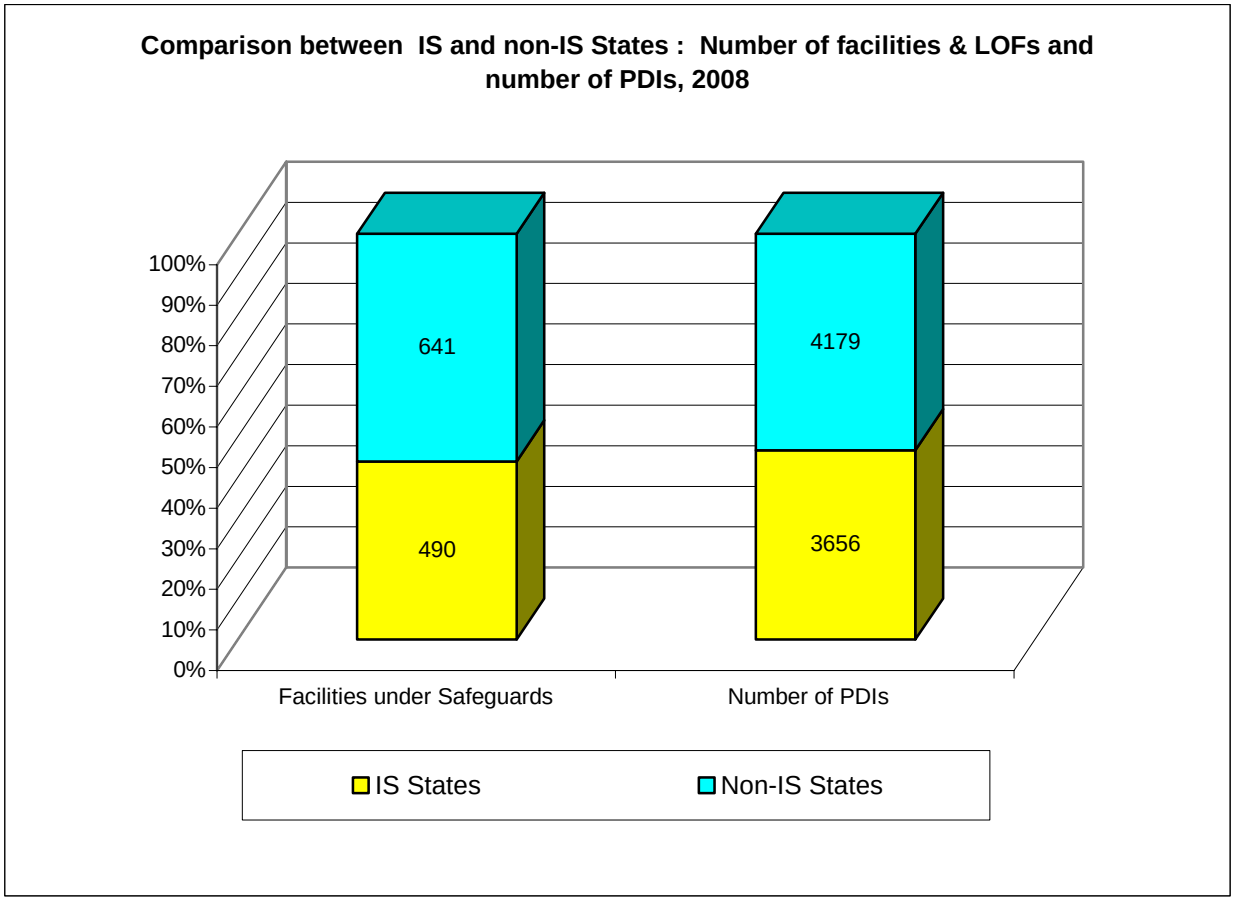


3. Implementation and Evaluation of Integrated Safeguards from 2001 to 2008.

The implementation of IS began in 2001 with Australia; in 2002 IS started in Norway and in 2003 in Indonesia. In 2004, 2005, 2006, 2007 and 2008, respectively 3, 7, 9, 14 and 25 States were under IS for the complete year. In 2008, IS was implemented during the year for eight additional States. In 2009, IS will be implemented for the complete year for 36 States.



With regard to the increase of States under IS, the total number of facilities and LOFs within the States for which IS was implemented in 2008 shows that this number is approaching the number of facilities and LOFs within States without IS implementation (considering CSA States with or without an AP in force). The inspection effort, reflected in person-days of inspections (PDIs), is getting closer between the two categories and both categories represent 95 % of the total inspection effort of the Secretariat. The figure below illustrates the status reached in 2008.



In 2008, integrated safeguards were being implemented for the entire year for 25 States (and Taiwan, China): in this group, 21 States (and Taiwan, China) have significant nuclear activities where safeguards implementation activities were carried out in accordance with the SLA and the AIP approved for each individual State; three other States have modified small quantities protocols (SQPs) in force; the 25th State has no SQP in force.

The activities carried out in 2008 included evaluation at Headquarters of all safeguards relevant information and verification activities carried out in the field:

(a) Headquarters activities relevant to objectives A, B and C:
- An SER for each of the 25 States (and Taiwan, China) under IS was updated and reviewed by the IRC.
- Evaluations of State accounting reports were carried out for all facilities and LOFs.
- Material balance evaluations were carried out for all facilities handling 1 SQ or more of nuclear material; for those of them handling material in bulk form, statistical analysis of material unaccounted for (MUF), shipper-receiver differences (SRD) and their cumulative values was performed.
- AP declarations were received and evaluated.

(b) In-field activities relevant to Objective A:
- CAs were performed at sites and locations declared under Article 2 of the relevant APs.
- During CAs, environmental samples and destructive analysis (DA) samples were taken.

(c) Inspection activities relevant to Objective B:
- DIVs were carried out in conjunction with inspections and at some facilities under construction or in the decommissioning phase.
- At research and power reactors capable of producing significant amounts of plutonium, containment/surveillance (C/S) and/or other unattended monitoring measures, complemented by unannounced or short-notice inspection regimes, were implemented at all these facilities.

- At enrichment, reprocessing and associated conversion facilities, C/S and/or other unattended monitoring measures complemented by continuous, regular or random inspection regimes were implemented. Limited frequency unannounced access (LFUA) was performed at the enrichment plants.
- Environmental samples were taken during inspections.

(d) Inspection activities relevant to Objective C:
- Inspections were carried out including physical inventory verifications (PIVs), unannounced inspections, random interim inspections and short notice random inspections (SNRIs). PIVs were performed in 127 of the 148 facilities handling one significant quantity (SQ) or more of nuclear material. For the facilities with no PIVs, State declarations were indirectly confirmed through implementing a random selection approach for a group of facilities or implementing an SNRI regime and taking into account verification results at other facilities in the group.
- Facilities handling unirradiated direct-use material were subject to regular visit for timely detection purposes.

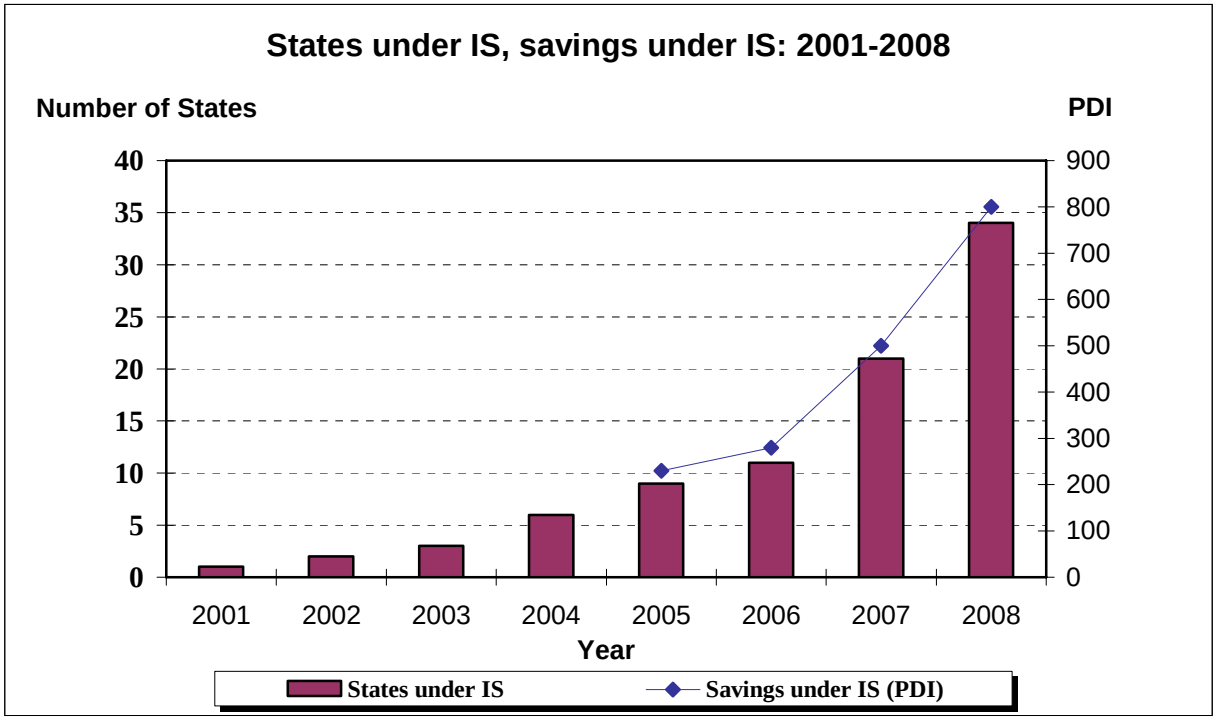
The IAEA concluded that the evaluation and verification activities performed in 2008 for the 25 States under integrated safeguards had been satisfactorily implemented and that the State-specific objectives had been achieved. Factors taken into account included the following:

- The results of some environmental samples and DA samples taken during inspections and CAs in 2008 were still outstanding at the time of evaluation due to delays in sample analysis/evaluation.
- There were delays of receipt of both accountancy reports and additional protocol declarations.
- Some SLAs needed revision or further development in order to continue improving the effectiveness and cost-efficiency of safeguards implementation, taking into account the acquired experience in the implementation of IS.

For 2008, the IAEA estimated that the implementation of IS resulted in saving of approximately 800 PDIs (for 2007: 500 PDIs). The savings are estimated for each State as the difference in PDIs between the year 2008 and the average inspection effort for the State before IS. A significant portion of those savings are related to the verification of transfers of spent fuel to dry storage. This saving is also due to the elimination of scheduled quarterly interim inspections for irradiated fuel (timeliness detection period extended from three to 12 months) and to the random selection of interim inspections and PIVs for groups of facilities. The implementation of remote monitoring systems is also reducing the number of interim inspections at facilities with unirradiated direct-use material.

The figure below shows the savings trend in relation to the number of States under IS. The first significant saving was for 230 PDIs in 2005, when IS implementation started gradually in Japan for a limited number of facility types. The next savings step was in 2007, 500 PDIs, when IS implementation started in Canada also in a limited number of facility types. A saving of about 300 PDIs is observed in Canada since 2007 with the implementation of unannounced inspection for transfer verification at multi-unit on-load refuelled reactors (OLRs) and their respective storages; this saving represents 40–50% of the inspection effort under traditional safeguards.

In Japan, for light water reactors (LWRs) and spent fuel storages, the saving is 50–60 % of the inspection effort under traditional safeguards and about 15% at RRCAs (research reactors and critical assemblies) and at DNLEUFFPs (natural and low enriched uranium conversion and fabrication plants.



While the figure above shows a reduction of inspection effort in the field, there was a substantial increase in activities at Headquarters related to the introduction of new facilities, evaluation of AP declarations, information analysis, including data being transmitted to the IAEA remotely, and State evaluations. This reflects the shift in the focus of safeguards implementation from verification of declared nuclear material at declared facilities to an information driven system that aims at understanding and assessing the consistency of information on a State's nuclear programme as a whole.

4. Integrated Safeguards Conclusions

Having evaluated IS implementation results, the IAEA concluded that there was no indication of undeclared nuclear material and activities and no indication of diversion of declared nuclear material from peaceful use in the 25 States under IS. On this basis, the IAEA concluded that, for these 25 States, all nuclear material remained in peaceful activities.

In addition, the IAEA was able to draw this conclusion using fewer in-field verification resources than would have been required if an IS approach had not been implemented. The scope for savings in in-field verification activities is greater for States with large, developed nuclear fuel cycles.

The State-level safeguards evaluation process needs further refinement. The envisaged changes would result in a 'living' document for each State, updated throughout the year, to match the AIP timetable. The focus would be on the evaluation of inspections, CAs and DIV against the SLTOs, providing continuous evaluation as AIP activities are performed.

One of the IAEA's goals in improving the effectiveness and efficiency of the safeguards system is to implement IS in all States with CSAs and APs in force.

Can the Additional Protocols be integrated into the European safeguards system?

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Abstract:

The safeguards regime in the European Union is set by the Euratom Treaty and the Safeguards Agreements with the IAEA (INFCIRC/193, INFCIRC/263 and INFCIRC/290). European legislation ensures the provision to the IAEA of data as required under the Safeguards Agreements. The Additional Protocol became a part of the legal bases for the safeguards in the European Union in 2004, and its reporting requirements were – partly – incorporated into the new European legislation (Euratom Regulation 302/2005). Since 2004 the Union has experienced two enlargements, with the gradual accession of the new Member States to the European Safeguards Agreement INFCIRC/193 and its Protocols.

This paper presents a review of some challenges to the European safeguards system that have their origin in the Additional Protocol. Can the AP requirements be simply added on to the framework provided by the Euratom Treaty? Can the European system cope with the new requirements stemming from the strengthened international safeguards system without loosing its original spirit? Can the European system benefit from the additional information and the additional activities stemming from the AP? These questions are addressed with the background of the first years' AP implementation experience within the European safeguards system.

Keywords: Additional Protocol, European Union

1. Introduction

1.1. Legal framework

The framework for the safeguards system in the European Union is laid down in Chapter VII of the Euratom Treaty. Its scope is stipulated in Article 77 as follows:

> *In accordance with the provisions of this Chapter, the Commission shall satisfy itself that, in the territories of Member States*
> *a) ores, source materials and special fissile materials are not diverted from their intended uses as declared by the users,*
> *b) provisions relating to supply and any particular safeguarding obligations assumed by the Community under an agreement concluded with a third State or an international organisation are complied with.*

The implementation of item b) includes the three Safeguards Agreements concluded with the International Atomic Energy Agency (IAEA) and the Member States (INFCIRC/193, 263 and 290). Some provisions in agreements concluded between the European Atomic Energy Community and third States (currently Canada, the United States of America, Australia, Japan, Kazakhstan, Ukraine and Uzbekistan) also fall under item b) [1].

Chapter VII of the Euratom Treaty and the three Safeguards Agreements with the IAEA can be seen as the two main pillars of the European Safeguards system. The Euratom Regulation is the piece of Community legislation that is foreseen in Article 79 of the Treaty, providing for the detailed implementation of both the Community safeguards and the requirements stemming from the Safeguards Agreements with the IAEA [2].

1.2. The question

This paper reviews and discusses the challenges ensuing from the advent of the Additional Protocols (AP) on the European safeguards scene. Can the provisions of the Additional Protocols be introduced and integrated into the European system? On a superficial level this question can be seen as enquiring whether and how the requirements stemming from the Additional Protocols could be incorporated into the European safeguards system so as to enable the Community (the European Commission) to produce the services it is expected to provide under the new provisions.

Alternatively the question can be read as asking whether such integration could happen without altering the original spirit of the European system. Could the essential elements of the European system be maintained with the Additional Protocols being implemented?

Yet another way of reading the question would be whether the Additional Protocols can be made an (integral) part of the Euratom safeguards system, so that the system, apart from fulfilling its obligations in the new environment, could also benefit from the changes induced by the Additional Protocols? Can the information stemming from the implementation of the Additional Protocols be used better, spurring the European system to evolve towards a full-scale system of regional safeguards? What follows will try to give elements for responding to these questions based on the implementation experience of the first five years.

2. Key elements of the European system

The safeguards system founded in the Euratom Treaty in 1957 was a European system as opposed to a national one in the original six Member States. For the first two decades of safeguards in Europe, the players were the holders of nuclear materials on the one side and the Commission as a control authority on the other. The Commission collected information from the holders and made inspections in their installations. Most Member States had in normal circumstances the role of a spectator: they received copies of information from the main players, but were called to intervene only in exceptional circumstances.

This setup did not change with the conclusion of the Safeguards Agreements between the Member States, the Community and the IAEA in late 1970s. The IAEA now started receiving from the Commission information provided by the holders of nuclear material, and the IAEA joined the inspections made by the Commission. The Member States´ role in routine implementation remained the same as before. In particular, any discussions concerning the implementation of the Safeguards Agreements were conducted between the Commission and the IAEA. Changes to this arrangement came more than two decades later, with the implementation of the Additional Protocols, as a new player – the Member State – entered the game.

The European nature of the Commission's mandate is reflected in the organisational arrangements within the Commission. Still today the implementation of the Euratom Treaty and the Safeguards Agreements (without the Additional Protocols) is – for inspection purposes at least – organised based on the type of installation, rather than the State where the installation is located. This organisational arrangement is expected to contribute to equality of treatment between similar installations located in different Member States. Aspects of State-based arrangement exist, however, especially in the accountancy sector.

Another element of the European system was – until recently – the fact that all inspections in the non-nuclear weapon States of the European Union were planned as common inspections for the IAEA and the Commission. This rule, enshrined in Article 14 of the Protocol to INFCIRC/193, still applies in the great majority of cases. In the past, the IAEA sometimes chose not to attend a planned inspection, but the Commission was always present. However, the Commission has recently decided not to maintain its presence as regards some inspections in given Member States. It is perhaps worth emphasizing

that this decision is independent of any requirement in the Additional Protocol, as misunderstandings sometimes arise in this regard.

3. Key elements of the Additional Protocols

The model Additional Protocol was adopted in 1997, and the Additional Protocols to the three European Safeguards Agreements were concluded in 1998 and entered into force in 2004 [3]. In terms of the Parties' involvement in routine implementation, the result was quite complex and different from the Safeguards Agreements. Whereas all reporting had previously gone from the holders of nuclear material to the Commission and further to the IAEA, now three different constellations were laid down, depending of the subject matter of the report (see Chart 1). Moreover, an explicit provision was created to the Additional Protocol to INFCIRC/193, permitting yet another reporting arrangement. By making a side-letter to the Agreement, the States could entrust to the Commission the implementation of some provisions that are under the Additional Protocol the State's responsibility.

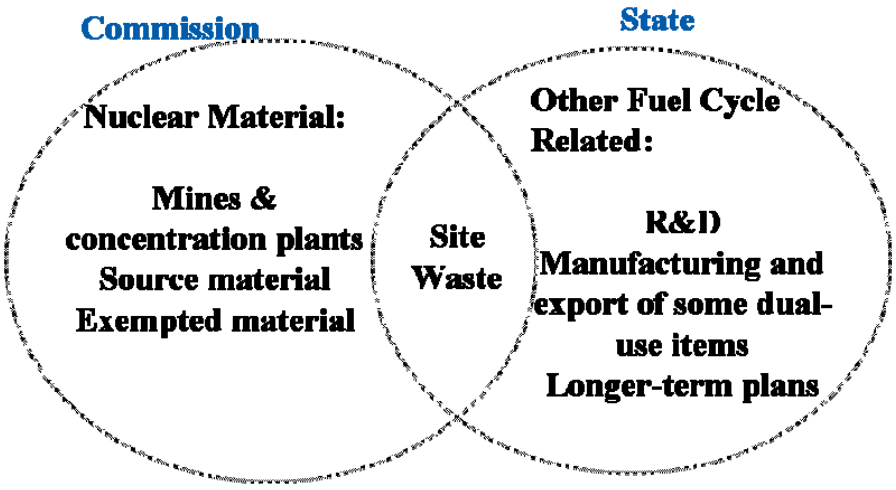

Chart 1: Reporting responsibilities under INFCIRC/193/Add.8.

3.1. Legal implementation

In order to enforce the new requirements generated by the Additional Protocols some legislative changes were needed. For the Community the new provisions were incorporated into the new Euratom Regulation 302/2005. The changes were essentially related to the collection of information on the sites under Article 2.a.(iii), and certain types of waste covered by Article 2.a.(viii). Other requirements concerning the provisions not involving nuclear material were incorporated into each Member State's national legislation.

For the provision of information under these Articles joint responsibility is laid down in the Additional Protocol. On charging the Commission to sign the Additional Protocol to INFCIRC/193 the Council explicitly stated that the final declaration under Article 2.a.(iii) was to be provided by the Commission. A site usually contains both buildings with nuclear material and buildings without. To ensure that complete declarations would be available to the Commission, the 'function' of Site representative was introduced. The Site representative was defined to be a person who would have to submit the site declaration and its updates to the Commission. The Site representative was not made responsible for the content of data. Instead it was explicitly laid down in Article 3(2) of the Euratom Regulation that the responsibility for the correctness and completeness of the information remained with the person or the undertaking operating the installation. Regarding buildings that do not contain nuclear material, the responsibility is attributed to the Member State.

Information permitting the Commission to fulfil its reporting obligations under Articles 2.a.(v), 2.a.(vi) and 2.a.(vii) existed already in the earlier versions of the Euratom Regulation. They were, however, further streamlined in the new Regulation.

The reporting requirements concerning nuclear fuel cycle related non-nuclear materials or equipment (Articles 2.a.(iv) and 2.a.(ix)), or those relating to applied R&D under Articles 2.a.(i) or 2.b.(i) were not included in the Regulation, as these do not involve nuclear material. The same is true for the State's plans for the development of nuclear fuel cycle (Article 2.a.(x)). While reporting under Article 2.a.(ix) is not included in the Euratom Regulation, there is European legislation covering the substance of the reporting requirements. The items whose export or transfer between Member States must be reported to the IAEA according to the Additional Protocols are listed in Category 0 of Council Regulation setting up a Community regime for the control of exports of dual-use goods and technology [1]. The items are all subject to the application by national authorities of export/intra-Community transfer licensing procedures, and therefore the responsibility for the control of exports to third States and shipments to other Member States is on each Member State.

3.3. Variable geometry

The side-letter arrangement that was originally chosen by 10 of the then 13 States was not included in the Regulation. It was necessary to create a basis on which the Commission could collect the information concerning the Articles not involving nuclear material. These were included in the same document (AP Implementation Arrangements) that was also to clarify the implementation process (timetable and responsibilities) with each Member State. While the detailed arrangements vary from State to State, in most cases the side-letter means that reporting is direct to the Commission from entities concerned with the obligation. For the AP Articles not involving nuclear material this is in practice implemented so that the State concerned would provide – and keep up-to-date – a list of entities covered by these reporting requirements. The Commission would ensure the collection of the necessary information from the listed entities. In some cases, however, a side-letter State preferred an indirect way of reporting: the national contact point would collect centrally all AP-specific information and provide it to the Commission. These different reporting channels represent quite some 'variable geometry' – a term once used in EU jargon to refer to different rules applying for different Member States. The channelling of information has been accommodated into the processes. In any case, all information provided under the Additional Protocol to INFCIRC/193 by one of the parties to another one (the Commission, a Member State or the IAEA) is copied to the third one.

4. Integrating the Additional Protocols

In this section we will look into some topics linked to making the Additional Protocols a part of the European safeguards system. The possibilities and current practice as regards Commission organisation and use of data originating from the AP requirements are discussed. The follow-up of installation status until decommissioning and the attitude towards small holders are taken as examples of areas where the Additional Protocols have had some impact on the European implementation praxis of safeguards.

4.1. Organisation

A sector was charged in the Commission with preparing the implementation of the Additional Protocols prior to their entry into force. The organisational arrangement remains the same. A group within DG TREN Unit H3 is in charge of the management of all data strictly linked to the Additional Protocols (receipt, preparation, sending and storing of declarations and other documents). Apart from this, the Group has the primary responsibility of managing all Complementary Accesses and ensuring the Commission's presence where necessary. The AP Group does not have tasks other than in the implementation of the AP. Its existence is not recognised in formal internal procedures.

The data generated or collected under the provisions of the Additional Protocols are stored in a separate database that was only recently made accessible to nuclear inspectors, carrying out other safeguards tasks. Its relative isolation keeps the implementation of the Additional Protocols largely outside the rest of safeguards work, including from internal routine information flows and decision-shaping.

Is this isolation an unavoidable consequence of the Additional Protocols? Did the safeguards organisation have to become more complicated, or is the increased complication due to a more or less deliberate choice made by the Commission itself? The Additional Protocols require reporting to be

based on States. Does this imply that the Commission organisation should move away from the traditional, based on installation types? No, it does not. The installation-based organisation, which is surely valid in an integrating European Union, can be maintained, while satisfying the requirements for State-based reporting. The Additional Protocol dimension can be added to the installation-based organisation quite easily, in the form of a light layer of coordination along the State-dimension. This complementary layer would cater for the implementation differences between the Member States.

4.2. Use of AP information

The Additional Protocols generate a fair amount of information complementary to the 'traditional' safeguards data. The Commission stores all declarations provided to the IAEA under the Additional Protocol to INFCIRC/193. Could this information be used to the benefit of the European safeguards system?

Starting from the daily work, one will easily identify nuclear inspections as possible beneficiaries of the data collected. The information included in the site declarations can be useful as a complement to the other information available for the preparation of inspections. It provides general descriptions of the buildings constituting the location where the installation is found, including a map. This type of information can be useful especially in the new era of short-notice inspections, where the inspector departing on inspection does not always have very detailed knowledge of the target location. The declarations are now available on-line for the inspectors needing them. Conversely, those preparing the final versions of the site declarations in the Commission, as the responsible entity for the nuclear part of the site declarations, could benefit from the return of information from inspectors, typically in cases where an inspector notices a need to update information or has other suggestions for the improvement of the data given. Such exchange of data could enhance results at both ends.

The implementation of the agreements with third States brings in data on nuclear fuel cycle related equipment, in addition to nuclear materials. Some of these data might be usefully employed in the implementation of the Additional Protocols and vice versa.

4.3. Small holders and the end of life of nuclear installations

The advent of the Additional Protocol with its requirement that all holders of nuclear material would need to be declared as 'sites' led to some new thinking in safeguards implementation, mainly as regards small holders (i.e. locations outside facilities (LOFs) without nuclear-fuel cycle related activities) and installations that in reality no longer existed as nuclear installations due to decommissioning or closing down.

4.3.1. Decommissioned and closed-down installations

The Additional Protocol highlighted the concept of 'decommissioned' installation in safeguards. Only once a facility was decommissioned, no site declaration would need to be provided. The status of installations after the removal of all nuclear material had in the past not been followed rigorously. As many installations that had been closed down for years now became subject to strengthened safeguards, there was an incentive for the European safeguards system to verify their true status and, where applicable, have them confirmed as decommissioned.

The Additional Protocols acting as a trigger, the Commission and the IAEA took interest in the end of life of nuclear installations. Criteria for performing practical status verifications were developed with the IAEA. As a result, over hundred verifications of installation status leading to the confirmation by the IAEA of the status as decommissioned were made between 2003 and 2008.

Some of these verifications took place even in small non-nuclear installations, where the meaning of 'decommissioning' is questionable. The work was probably partly driven rather by ad hoc requests than mature reflection. The safeguards reporting requirements may have had their influence, as well. Once a facility has been confirmed as decommissioned, all its reporting obligations cease towards the IAEA and the European system of safeguards. For LOFs this happens already once the installation has been confirmed as closed down, with the exception of those that are part of an AP site and therefore remain subject to reporting requirements under the Additional Protocol. With regard to the IAEA, the situation is different insofar as the need for reporting for nuclear material accountancy

(NMA) purposes even after closing down was recently introduced (zero PIL (physical inventory listing) required by the IAEA). This seems to have led to excessive zeal in decommissioning.

All in all, one can say that the Additional Protocol gave a push towards more rigour in the European safeguards system. The life-cycle of nuclear installations started to be followed to the end, progressively reducing the number of installations in the list of installations under the Commission's control.

4.3.2. Lighter safeguards for small holders?

Another example of a change of policy of sorts is the handling of small holders with nuclear material. Before the Additional Protocols the Commission had not, except in very rare cases, used the possibility of exempting nuclear material from the IAEA safeguards. Practically all nuclear material had been kept subject to the requirement of inspections and monthly reporting. Although all small holders were subject to full safeguards, the controls on them were in practice loose, as the reporting requirements were not strictly enforced for these holders that mostly had material and activities of low proliferation relevance.

The Additional Protocol triggered a need to change, as all holders with non-exempted nuclear material were required to be declared as AP sites. The wish to avoid reporting hundreds of small holders as sites led to requests that the Commission change its practice and start using exemption possibility as a way of avoiding a proliferation of sites in the European Union. Exempting material in LOFs became the declared policy for the Commission on the eve of the entry in to force of the Additional Protocols.

Legislative steps were taken towards lighter safeguards for small installations in the Euratom Regulation. New provisions for reduced reporting were introduced into the Regulation under the concept of 'derogation', permitting holders of nuclear material to ask for limited frequency reporting (once a year, instead of monthly). The conditions for derogation were similar – although not exactly the same – as those laid down in the Safeguards Agreement for exemptions from IAEA safeguards.

The use of derogations and exemptions has had a hesitant start in the Commission. Exemptions can be used – as is done today – on a case-by-case basis for the specific purpose of avoiding site declarations and Complementary Access linked to sites. The cases of exemption have been few and the processes for obtaining and managing them have not been developed to routine. Exemptions and derogations, employed together or separately, could be important elements in a general framework of lighter safeguards applicable for small installations with little proliferation relevance Setting up such framework would, first of all, require an overall evaluation of the different safeguards options for small holders, followed by decisions on policy reorientation and the development of working practices needed for implementation. While these steps could be best initiated by the Commission, the focal point in the nuclear material accountancy and control (NMAC) in the European Union, all other stakeholders would need to be involved in the developments, as well. The holders of nuclear material would need to be brought to adopting appropriate NMA practices. In this context the Commission should enable frictionless reporting by offering guidance and adequate tools, while strictly enforcing the rules. The Commission could also take initiatives towards enhancing the implementation by the IAEA of the exemption rules enshrined in the Safeguards Agreement.

Increases in the effectiveness and efficiency of safeguards with small holders could be attainable as a result of exploring the potential and putting it into use with determination. Good outcomes in future can be achieved only with an investment made today.

4.4. Involvement of Member States

As described above, the implementation of safeguards before the Additional Protocols was a game with two main players, the Commission and the IAEA, including nuclear operators reporting directly to the Commission. The Member States were not asked to actively contribute to it. In fact, the opposite happened. On each enlargement until 1995, a downsizing in the national safeguards organisation took place, as tasks were taken over by the Commission.

With the Additional Protocols, the Member States got an active role, and contacts with the national contact point in each Member State became a routine matter for the AP Group, even where

implementation work was being carried out by the Commission. The principle of variable geometry (section 3.3.) applies, but is no hindrance to cooperation. Would it be useful to try and integrate this novelty into the European system, the Member States thus becoming partners in the European safeguards implementation? Could this development help the European system to remain truly functional in the new framework conditions? The Additional Protocol as a herald to changing the mindset in the European system?

When looking into the practice, one can already see greater active involvement of some Member States in the implementation of European safeguards, particularly in the new Member States. In some of them the Member State national authority has taken the task of managing the NMA reporting for the small holders. Even in others the national authority is facilitating the reporting for the small holders, while the legal responsibilities remain with each holder. By these means it has been possible to avoid language barriers and receive better reporting than otherwise. One might ask why the same principle could not apply to small holders in other States where a national authority is already collecting accountancy information. Would it not be beneficial for the European Safeguards system as a whole to use the existing resources and local knowledge in respect of those holders that require disproportionate effort compared to the proliferation risk they represent?

A concept coined in the general European integration comes handy – subsidiarity – meaning that responsibilities should be at the lowest level of government where they sensibly can be. An extension of this principle could lead to some tasks of the European safeguards system being taken care of at the level of the Member State or even regional authority. The overall legal responsibility for safeguards in the European Union would obviously not be affected by such arrangements.

4.5. Complementary Access

The verifications under the Additional Protocols (Complementary Access) are implemented by the Commission as pure IAEA activity. The role of the Euratom inspector is to accompany the IAEA, acting as an observer and facilitator where needed. As a result of its presence in the Complementary Access, the Commission obtains information that gives a fairly comprehensive overview of the use the IAEA makes of its new verification powers in Europe.

So far little thought has been devoted within the Commission to whether and how the presence of Euratom inspector could be used for purposes of the European safeguards system itself. Could the inspector collect information from the location or make verifications for the European system? What could be done and under which circumstances, taking into account the IAEA's goals for Complementary Access? The Agency's practice of announcing Complementary Access practically always with the minimum notice time (24 hours vs. 2 hours in advance) would probably need to change, in order to give some time for preparation to the Euratom inspector. This should not be impossible, given that the current practice is not fully in line with the letter and the spirit of the Additional Protocols, or with the internal implementation guidelines.

5. Conclusion: back to the original spirit

We have seen that the advent of the Additional Protocols has brought some challenges to the European safeguards system, both in terms of legal environment and implementation practice. Changes in the legislative framework were adopted and are now part of the European system. Adapting the implementation practice so as to benefit from the new legislative features is still far from being achieved. Indeed, due to a combination of organisational aspects and factors related to tradition and managing change, little drive has been observed towards greater integration of new elements into the System.

The 'European spirit' of the Commission at the Community level providing assurance of nuclear material not being diverted from its declared use can be maintained and enhanced, probably not in exactly the same form as in the past, but adapted into the current international environment. This will require re-thinking of organisation and its functioning. Policy and work processes in some areas need to be adapted, ensuring better coordination. Promoting a new relationship with the Member States can also prove fruitful. Learning to use for European safeguards purposes the data stemming from reporting under the Additional Protocol will make European safeguards stronger.

The Additional Protocol can be integrated into the European safeguards system. More than that: it could serve as a trigger, giving the European system an impulse to develop into an integrated regional system of safeguards. Should there be a political will, this system could carry the responsibility of safeguards tasks regionally, acting as a part of the world-wide safeguards system.

[1] Synetos S, MacLean F, Hoeke J: *Euratom Co-operation Agreements, Safeguards, and Export Controls*. ESARDA Symposium on Safeguards and Nuclear Material Management, 31st Annual Meeting, 26-29 May 2009. Vilnius.

[2] European Commission; *Commission Regulation (Euratom) No 302/2005 of 8 February 2005 on the application of Euratom safeguards;* OJ L 54, 28.2.2005, p. 1.

[3] INFCIRC/193/Add.8, INFCIRC/263/Add.1 and INFCIRC/290/Add.1.

[4] The revised regulation was adopted on 5/5/2009 by the Council of the European Union under the title *Council Regulation setting up a Community regime for the control of exports of dual-use goods and technology*. The revised regulation had not been published at the time of writing.

On the Distribution of Unannounced Interim Inspections in Time - A Hybrid-sequential Approach -

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Abstract:

Unannounced interim inspections of nuclear facilities in the European Union have been discussed already for some time by EURATOM and IAEA experts. Therefore, and since they pose challenging mathematical problems, they have been analyzed in detail in the framework of a project carried through in the Institute for the Protection and Security of the Citizen of the Joint Research Centre (JRC-IPSC), located in Ispra, in collaboration with ITIS e.V. an der Universität der Bundeswehr München.
A game theoretical study is performed under the assumption that the safeguards inspector wants to minimize the time between the start of an illegal activity – if there is any – and its detection. Also, it is assumed that the plant operator takes his decisions sequentially, i.e. he decides at the beginning of the inspection period (in general one year) only if to start an illegal activity immediately or not, in the latter case again after the first inspection and so on. Of course more assumptions are made in order to specify the model. As a result of the analysis, among others optimal inspection strategies in a facility and the expected detection times in nuclear facilities depending on the total number of interim inspections in a facility, and the error of the second kind probability are determined, and the trade-off between these two parameters is discussed.

Keywords: Unannounced Inspections, Game Theory, IAEA, EURATOM

1. Introduction

The appropriate number and timing of unannounced interim inspections in nuclear facilities in the framework of nuclear material safeguards has been discussed in the safeguards community already for a long time and for good reasons. The matter is relevant both for EURATOM and IAEA safeguards authorities.

For IAEA safeguards, the implementation is presently shifting from a system mostly focused on traditional safeguards to the so called "Integrated Safeguards", where the verification system is more holistic and State level based. Consequently, at least in some cases (i.e. for some facilities in some States), there will be a decrease in the yearly number of fixed scheduled interim inspections by substituting some of them with unannounced ones.

For EURATOM safeguards there is also an evolution of the way to implement inspections in EU together with IAEA activities. Most of the IAEA inspections in EU will continue to be carried out in presence of EURATOM inspectors. On an other side, like the IAEA, EURATOM may also carry out unannounced inspection by its own.

The analysis of this general problem is in some cases mathematically demanding, and concrete solutions, i.e., advices on numbers and points of time for specific facilities as well as effectiveness and efficiency considerations, depend crucially on special modelling assumptions. Therefore, a joint project was carried through by the Institute for the Protection and Security of the Citizen of the Joint Research

Centre (JRC-IPSC) in collaboration with ITIS e.V. an der Universität der Bundeswehr München in the course of which the assumptions necessary for a quantitative analysis were carefully collected and classified. In the following we will refer to the project report [1] for details which cannot be presented here for space reasons.

The most important assumptions to be made are the following ones:

- *Planning:* Whether or not the inspections have to be planned at the beginning of the reference time interval, e.g., one calendar year, and whether or not they can be observed by the facility operators;
- *Time:* If inspections can take place at any points of time (continuous time), or only at discrete ones (The meaning of these alternatives will be explained in the third section.);
- *Inspection philosophy:* If the safeguards authority uses a concept of the kind "the earlier an illegal activity is detected the better" (*playing for time*) or that any illegal activity has to be detected "within a specific time" (*critical time*);
- *Sampling:* If statistical errors of the first (false alarms) and second kind (failing to detect illegal activities) have to be taken into account.

In Figure 1 a classification of these assumptions is represented graphically. α and β are the error of the first and second kind probabilities.

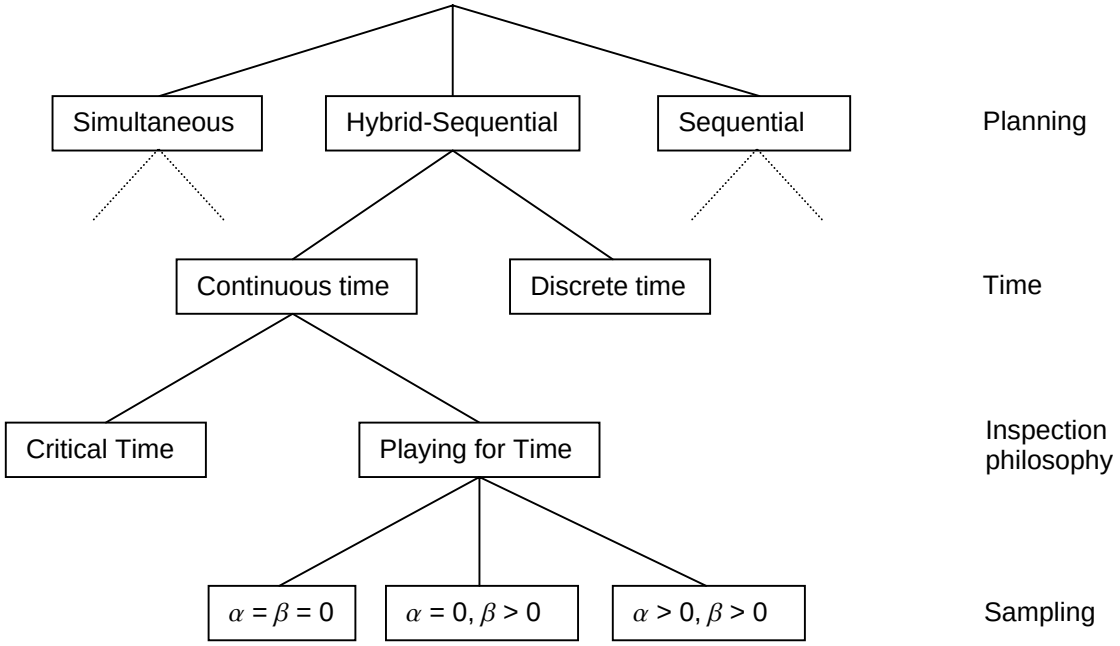

Figure 1: Classification of assumptions.

Of course, it is neither possible with reasonable effort, nor interesting from a practical point of view to consider 36 models which require in part different analytical and numerical techniques. Instead, in the course of the afore mentioned study [1] four of these 36 models, namely simultaneous and hybrid-sequential playing for time games, both time discrete and continuous, were selected and analyzed. The results were applied to inspections in two prototypical types of nuclear facilities, namely an on-site spent fuel storage facility and an fuel element fabrication facility employing Low Enriched Uranium. These two cases correspond to an "item" facility, i.e. a facility were nuclear material in handled in item form only and a "bulk" handling facility, where the material is handled both in item as well as in bulk form.

In the following, for good reasons which will be explained at the end of the third section, one special case, namely the hybrid-sequential-continuous time playing for time model, will be selected, and its applications to EURATOM and IAEA inspections in nuclear facilities of States of the European Union will be discussed. In the second section this model will be presented and analyzed. In the third section the practice of interim inspections in on-site interim storage facilities will be described, and the findings of the second section will be applied to one single on-site interim storage facility. The fourth section contains an outlook to the problem of determining the appropriate number of unannounced interim inspections in other facilities and in one State of the European Union.

2. Game Theoretical Model

Quite generally, let us assume that in a facility *k* unannounced interim inspections will be performed in a reference time interval $[t_{k+1}, t_0]$, see Figure 2, at the beginning and end of which a physical inventory verification (PIV) is performed. The backward counting simplifies the mathematical analysis and the presentation of the solutions, also the use of t_{k+1} instead of zero. t_0 is determined by the absolute length and scaling of the reference time interval. If this interval is one year, and time is measured in quarters of years, e.g., then we get $t_0 = 4$.

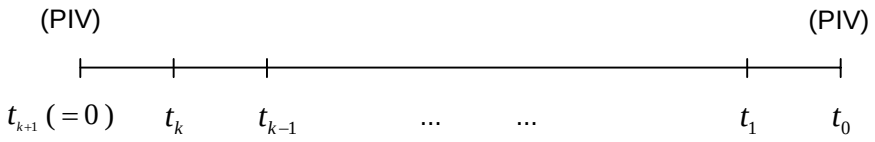

Figure 2: Time line of *k* interim inspections.

The operator of the facility will start the illegal activity within this time interval (Legal behaviour is also considered in [1]), and this illegal activity will be detected with probability $1-\beta$ by the next inspection, and, if not earlier, with certainty at t_0 , the time point of the physical inventory taking of the operators and of the Physical Inventory Verification (PIV) by the inspector. In our application, a generic on-site interim storage facility, we consider only attribute sampling inspections procedures (see [2]), therefore, errors of the first kind, i.e., false alarms, are excluded here. Whereas, the inspector decides at the beginning of the reference time interval when the *k* inspections shall take place, the operator chooses a sequential procedure, hence the name of hybrid-sequential model. At t_{k+1}, i.e., the beginning of the game, the operator decides only whether or not to start the illegal activity immediately. If not, he decides again after the first inspection at t_k whether or not to start the illegal activity immediately, and so forth, until t_1. The payoff to the operator is the time elapsed between the start and the detection of the illegal activity, and the payoff to the inspector is its negative value. In other words, we consider a zero sum game.

More assumptions have to be made. For the sake of completeness, we give them as a list.

- There are two players: operator and inspector.
- The inspector can perform his inspections at any time point between t_{k+1} and t_0.
- The operator behaves illegally.
- The number of interim inspections is also known to the operator. At most two unannounced interim inspections are permitted in one facility and within the reference time interval.
- The inspector decides at the beginning of the reference time interval when to perform his inspections. The operator decides at the beginning of the reference time interval whether to start his illegal activity immediately or only right after the inspection(s).
- We assume that the inspector may commit an error of the second kind, i.e., an illegal activity is not detected with probability β per inspection although there is one.

- The payoff to the operator is the time between the start of the illegal activity and its detection. The payoff to the inspector is the negative one (zero-sum game).
- In case of the coincidence of the start of the illegal activity and the inspection, the illegal activity is detected only at the occasion of the next inspection or at the PIV.

These assumptions, which are based on practical considerations and are justified extensively in [1], represent the basis for the mathematical models of unannounced interim inspections presented in the next sections.

2.1. One unannounced interim inspection (k=1)

As already mentioned, the operator decides at the beginning t_2 of the reference time interval $[t_2, t_0]$ whether to start his illegal activity immediately or only after the inspection. If not, he has to do this at point t_1. The graphical representation of this two-person zero-sum game in extensive form, see e.g. [3] and [4], is given in Figure 3. This kind of games is suited best to model information states of players in conflict situations.

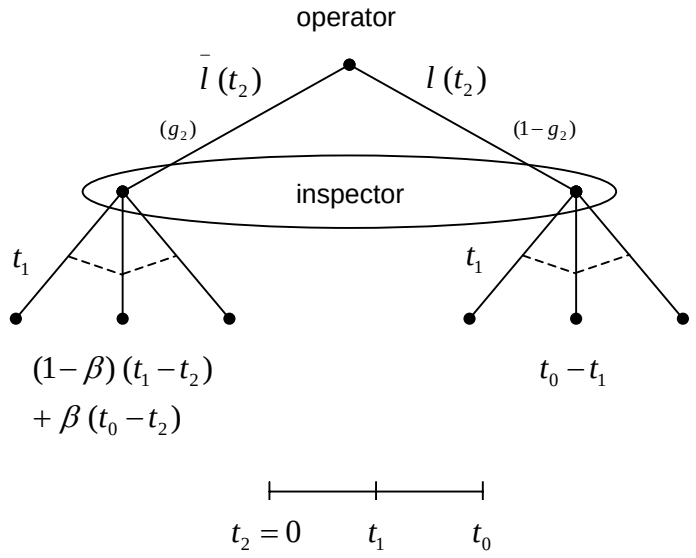

Figure 3: Extensive form of the continuous time hybrid-sequential inspection game with one interim inspection.

At the top of this figure it is indicated that at time point t_2 the operator either does start his illegal activity, $\bar{l}(t_2)$, or he does not, $l(t_2)$. The inspector chooses at t_2 a time point t_1 for inspection without knowing the operator's decision at t_2. This is indicated by the oval which is called the information set of the inspector.

If the operator chooses $\bar{l}(t_2)$ and the inspector performs his unannounced interim inspection at time point t_1, then the expected (conditional with respect β) detection time is given by

$$(1-\beta)(t_1-t_2)+\beta(t_0-t_2),$$

whereas in case the operator chooses $l(t_2)$ it is given by t_0-t_1.

Let g_2 be the so-called behavioural strategy of the operator, i.e., the probability to start the illegal activity at time point t_2 . Then the (unconditional) expected detection time is

$$Op_1(\beta; g_2, t_1) = g_2\,[\,(1-\beta)\,(t_1 - t_2) + \beta\,(t_0 - t_2)\,] + (1-g_2)\,(t_0 - t_1)\,.$$

The solution concept in game theory is based on the so-called Nash-equilibrium, see [5]: It is defined as that pair of strategies of both players that has the property that any unilateral deviation from that equilibrium does not improve the deviator's payoff. That is for the game discussed here: we are looking for a pair (g_2^*, t_1^*) with the property

$$Op_1(\beta; g_2, t_1^*) \le Op_1(\beta; g_2^*, t_1^*) \le Op_1(\beta; g_2^*, t_1)$$

for all $g_2 \in [0,1]$ and $t_1 \in [t_2, t_0]$. These strategies g_2^* and t_1^* are also called optimal strategies of the operator and the inspector.

The solution of this game is given as follows: The optimal inspection time point t_1^* is given by (remember $t_2 = 0$)

$$t_1^* - t_2 = \frac{1-\beta}{2-\beta}\,(t_0 - t_2) \tag{1}$$

and the optimal operator strategy by

$$g_2^* = \frac{1}{2-\beta}\,.$$

The optimal expected detection time is

$$Op_1^*(\beta) = Op_1(\beta; g_2^*, t_1^*) = t_0 - t_1^* = \frac{t_0 - t_2}{2-\beta}\,. \tag{2}$$

It should be emphasized that our analysis leads to an explicit dependence of the optimal time point for inspection t_1^* on β , see the left lower part of Figure 5 for $t_0 = 4$. Whereas for $\beta = 0$ the common sense point of view would lead to this result, for $\beta > 0$ one would hardly arrive at this result without quantitative analysis. The same holds for the operator's optimal strategy.

Also it is interesting to note that the optimal time point for inspection t_1^* depends on the length $(t_0 - t_2)$ of the reference time interval and on β, while the optimal strategy of the operator g_2^* is only a function of β. It is intuitive, however, that both t_1^* and g_2^* decrease with increasing β: for β close to 1 the detection probability is close to zero and therefore the operator starts his illegal activity with probability close to 1 at time point $t_2 = 0$. Consequently, the inspector will perform his inspections also very early.

Finally and most importantly, the optimal strategy of the inspector is a *pure* strategy, i.e., t_1^* is deterministic. In other words, the inspector can announce the time point of his interim inspection if he wishes so (and which the operator knows anyhow). It can be shown, see [1], that the inspector can also choose the time point t_1 for inspection using an arbitrary distribution density $f(t_1)$ concentrated on $[t_2, t_0]$ such that his optimal expected time point t_1^* for inspection,

$$t_1^* = \int_{t_2}^{t_0} t_1\; f(t_1)\, dt_1$$

is the same as the deterministic one given by (1). However, this way he does not gain anything.

2.2. Two unannounced interim inspections (k=2)

Again the operator decides at the beginning t_3 of the reference time interval $[t_3, t_0]$ whether to start his illegal activity immediately or later. In the latter case he decides after the first inspection at t_2, and finally if he does not start the illegal activity at t_2, he has to do this at t_1.

The extensive form of this game would have to show this information structure. Unfortunately, it is difficult to be represented and does not provide real insight. Furthermore, the expected detection time can be written down in a straightforward way. For all these reasons we do not show the graphical representation of this game.

Let (g_3, g_2) be the behavioural strategy of the operator, i.e., the probabilities to start the illegal activity at time point t_3 or t_2 .Then the (unconditional) expected detection time is

$$\begin{aligned} Op_2(\beta;(g_3,g_2),(t_2,t_1)) &= g_3\,[\,(1-\beta)\,(t_2-t_3)+\beta\,(1-\beta)\,(t_1-t_3)+\beta^2\,(t_0-t_3)\,] \\ &+ (1-g_3)\,[\,g_2\,((1-\beta)\,(t_1-t_2)+\beta\,(t_0-t_2))+(1-g_2)\,(t_0-t_1)\,]. \end{aligned}$$

Again we are looking for an optimal strategy (g_3^*, g_2^*) of the operator and an optimal strategy (t_2^*, t_1^*) of the inspector fulfilling the inequalities

$$Op_2(\beta;(g_3,g_2),(t_2^*,t_1^*)) \le Op_2(\beta;(g_3^*,g_2^*),(t_2^*,t_1^*)) \le Op_2(\beta;(g_3^*,g_2^*),(t_2,t_1))$$

for all $(g_3, g_2) \in [0,1]\times[0,1]$ and all (t_2, t_1) with $t_3 < t_2 < t_1 < t_0$. The solution of this game is recursively given by (remember $t_3 = 0$)

$$t_2^* - t_3 = \frac{1-\beta}{3-2\,\beta}\,(t_0 - t_3) \qquad \text{and} \qquad t_1^* - t_2^* = \frac{1-\beta}{2-\beta}\,(t_0 - t_2^*) \tag{3}$$

(note that the second formula (3) is formally the same as (1)) as well as

$$g_3^* = \frac{1}{3-2\,\beta} \qquad \text{and} \qquad g_2^* = \frac{1}{2}.$$

The optimal expected detection time is

$$Op_2^*(\beta) = Op_2(\beta;(g_3^*,g_2^*),(t_2^*,t_1^*)) \;=\; t_0 - t_1^* = \frac{t_0 - t_3}{3-2\,\beta}\;. \tag{4}$$

Since $t_1^* = 2\,t_2^*$, we obtain that for $\beta < 1$ the second inspection takes place after the double the time than the first one. For $\beta = 0$ we get

$$t_2^* = \frac{1}{3}\,t_0 \qquad \text{and} \qquad t_1^* = 2\,t_2^* = \frac{2}{3}\,t_0\,.$$

As in the case $k = 1$, for $\beta = 0$ the common sense point of view would lead to this result, for $\beta > 0$ one would hardly arrive at this result without quantitative analysis. The same holds for the operator's optimal strategy (g_3^*, g_2^*): Since the operator is confronted at t_3 with three inspection intervals of

equal length he chooses $g_3^* = 1/3$. After the first inspection however, only two intervals of equal length are left. Thus, he chooses $g_2^* = 1/2$.

Again, most importantly is the fact, that the inspector may announce the optimal time points of his inspections, if he wishes so, and the same arguments as given in the previous case hold as well.

3. Applications

It was mentioned in the introduction that as a first example an on-site interim storage facility for spent nuclear fuel elements is chosen. The origin and function of this type of facilities were described by Behrens et. al. [6] and Rudolf et. al. [7]. The on-site interim storage of Emsland Nuclear Power Plant is hereafter chosen, as a typical representative of an on-site interim storage facility, although the considerations here presented can be extended to any facility of this type.

The facility consists of two buildings, namely storage building with storage area and reception area for spent fuel casks, and control building in which plant operations are controlled. The permitted storage period is limited to 40 years beginning with the emplacement of the first spent fuel cask in the storage building. There are 130 cask positions, five of which being reserved for empty casks only. The Lingen interim storage facility has a length of about 110 m, a width of about 30 m, and a height of about 20 m. The wall thickness is about 1.2 m, while the monolithic roof is about 1.3 m thick. The floor is made of concrete armoured with steel.

In the reactor containment, spent fuel elements will be loaded into shielding casks, e.g., of the CASTOR-type (cask for storage and transport of radioactive material), and then transferred out of the reactor building into the associated on-site storage facility.

It has been agreed between the German State authorities, EURATOM and the IAEA that in each of the on-site interim storage facilities once a year a physical inventory verification is done, and that every three months a routine inspection is performed. The main purpose of the routine inspections is to check the seals at the casks on a random sampling basis.

For the subsequent quantitative analysis we consider a representative situation where there are *N* casks (80 to 190) with spent LWR fuel elements in the storage facility, and where each cask contains 19 spent fuel elements, see [7]. Without going into the details of the usability of the plutonium (Pu) for weapons in the fuel elements, see, e.g., [8] or [9], we assume that there are about 5 kg Pu in each fuel element. Therefore, in order to illegally acquire one significant quantity[1], the seals of *at most two casks* need to be broken. Here it is assumed that the seal of at most one cask needs to be broken, which represents the worst case for the inspector. In other words, during one routine inspection one broken seal has to be detected with sufficient probability $1-\beta$.

Quite generally, let the total number of seals be *N*, the number of checked seals be *n*, and the number of broken seals be *r*. Then according to the hypergeometric distribution law, the probability to detect at least one broken seal in case of drawing without replacement is (see, e.g., [2])

$$1-\beta(N,n,r) = 1-\binom{r}{0}\binom{N-r}{n-0}\Big/\binom{N}{n}.$$

Thus, for the worst case from the side of the inspector $r=1$, i.e., only one seal is broken, we get

$$1-\beta(N,n,1) = \frac{n}{N}, \tag{5}$$

[1] The IAEA defines a significant quantity as “the approximate amount of material for which the possibility of manufacturing a nuclear explosive devise cannot be excluded. For Pu this is set to 8 kg.

which means that probability of detection is proportional to the number of checked seals.

According to the IAEA Safeguards Glossary [10], *Integrated Safeguards* (IS) is the optimum combination of all safeguards measures available to the IAEA under comprehensive safeguards agreements and additional protocols to achieve maximum effectiveness and efficiency in meeting the IAEA´s safeguards obligations within available resources. There is also the IAEA/EURATOM Partnership Approach (PA), which is an approach for implementing safeguards in the non-nuclear-weapon States of EURATOM. It updates to Integrated Safeguards (IS) the approach firstly agreed between the IAEA and EURATOM in 1992, see [11]. The partnership approach provides for common use of safeguards equipment, joint scheduling of inspections and special arrangements for inspection work and data sharing by the two organizations. The NPA enables the IAEA to economize on safeguards equipment and inspection efforts deployed in the relevant States while maintaining its ability to perform independent verification.

At present the traditional procedure is applied to on-site interim storages, i.e. both EURATOM and IEAE inspectors are present when the inventory is verified and when the three routine inspections per year are performed. In the framework of the above described concepts, IS and PA, it is discussed [11] that, in the future, only EURATOM inspectors perform all routine inspections, and that IAEA inspectors perform *Unannounced Interim Inspections*. At this point it should be mentioned that two types of interim inspections are planned, namely *inspections with short notification (SNRI)*, e.g., form one to a few days, and *unannounced interim inspections* with no advance notification. To model the difference between these two types it is necessary to make assumptions about the operator´s possibilities to camouflage illegal activities within the advanced notification time. Since definite decisions have not yet been made, we discuss two alternatives in the study [3], namely that IAEA inspectors join EURATOM inspectors while they perform routine inspections, or perform their unannounced interim inspections at any time, independent of the EURATOM inspections.

Figure 4 combines all findings of this paper and shows their dependencies and practical implications.

The optimal expected detection times are drawn in the upper diagram as functions of the non-detection probability β for the two cases *k=1* and *k=2*, see formulae (2) and (4). We have chosen $t_0 = 4$ according to the quarterly inspections of EURATOM which means that the optimal expected detection times are measured in quarters of years. Choosing for instance $t_0 = 12$ would lead to a measurement in months.

It can be seen that $Op_2(\beta;(g_3^*, g_2^*),(t_2^*, t_1^*)) \le Op_1(\beta; g_2^*, t_1^*)$, which is very intuitive, since the more unannounced interim inspection(s) are possible the shorter the expected detection time will become. In case of $\beta = 1$, i.e., the detection probability is zero, any illegal activity is detected only at the end of the reference period and therefore the detection time is 4.

If the desired optimal expected detection time is about 1.5 quarters of a year, we see that this optimal expected detection time cannot be reached with one unannounced interim inspection (*k=1*), since with (2) we get

$$Op_1^*(0) = Op_1(0; g_2^*, t_1^*) = 2 > 1.5\,.$$

The mid diagram in Figure 4 shows relation (5), i.e., the number of checked seals in case of *N=100,* the total number of seals as a function of β. It can be seen that the sample size *n* decreases with increasing β, which is intuitive.

The two lower diagrams present the optimal time point(s) for inspection(s) as given by formulae (1) and (3): on the left side for *k=1* and on the right side for *k=2*. It is interesting to note that for *k=1* and arbitrary β the optimal time point for inspection always lies between 0 and 2, whereas in case of *k=2* the first inspection time point lies between 0 and 2/3 where for the second time point we have 1/3 and 2/3 quarters of a year.

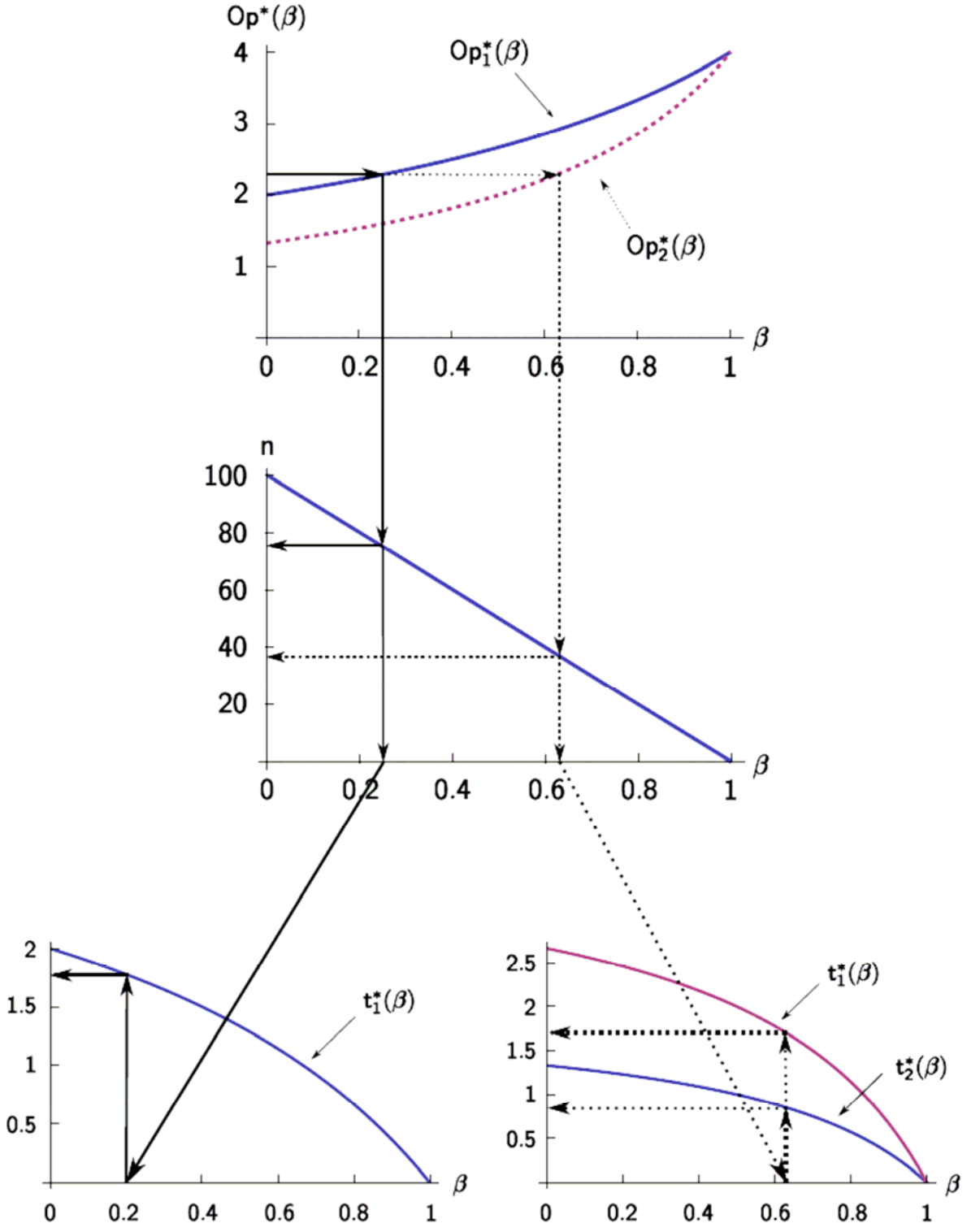

Figure 4: Graphical presentation of results.

All diagrams of Figure 4 can now be linked together as follows: Suppose the desired optimal expected detection time is about 2.25 quarters of a year. Then we see that

- In case of *one* unannounced interim inspection the non-detection probability has to be about 0.22 and therefore – using the mid diagram – the sample size has to be about 78. The corresponding optimal time point for the inspection is – using the left lower diagram – about 1.75 quarters of a year. The arrows with the solid lines illustrate this argumentation.

- In case of *two* unannounced interim inspection the non-detection probability has to be about 0.61 and therefore – using the mid diagram – the sample size has to be about 39 for each inspection. The corresponding optimal time points for the inspections are – using the right lower diagram – about 0.8 and 1.75 quarters of a year. The arrows with the dashed lines illustrate this argumentation.

We see that we can assure the same optimal expected detection time of 2.25 quarters of a year with one or two unannounced interim inspections. In both cases we have to check in total the same number of seals, namely 78, which follows from the formulae (2), (4) and (5). It depends on the overhead costs which case is more economic for the inspection authority.

There is a second way in which Figure 4 can be interpreted (although not indicated with arrows). Starting with the mid diagram we assume that we can only check a small number of seals *n*. Then we see that we arrive at a quite high non-detection probability β and therefore – using the upper diagram – at quite high optimal expected detection times.

As mentioned in the introduction we have been analyzing 4 models in [1]. Now the question arises, why we have chosen this model for this contribution and not one of the other ones. There are two reason: Firstly, the optimal solution of this game for the inspector is a *pure* strategy and therefore has the chance to become generally accepted. Secondly, this model gives the largest optimal expected detection time (under the 4 models considered in [1]) and is therefore the worst case for the inspector and can be taken as a reference model for planning inspections in a most conservative way.

In favour of the first alternative (IAEA and EURATOM pay joint visits) it should be mentioned that an additional burden is posed on the plant operators if IAEA inspectors visits the plant at different points of time than EURATOM inspectors.

4. Concluding Remarks

One of the purposes of the study was to discuss how many assumptions have to be made in order to arrive at the quantitative description of unannounced interim inspections. Among the many possibilities one concrete set of assumptions was presented here, which resulted in the game theoretical model given in the second section. Let us conclude with two complementary remarks.

First, in the game theoretical model facilities specific details are condensed in the parameter β, representing the non detection probability given that an illegal activity has been started. We have shown in the third section what its meaning is in the case of interim storages. In the study, see [1], this model has also been applied to a fuel fabrication facility and here, the meaning of β may be partially different. It should be mentioned in this context that if variable sampling procedures are used for the inspections, then errors of the first kind (false alarms) have to be taken into account.

Second, the number of unannounced interim inspections in the facility under consideration and for one reference time interval was also considered as a parameter of the game theoretical model. In fact, according to the lines of IAEA/EURATOM-Partnership Approach, see [11] and [12], that only the total number of unannounced interim inspections per year in one type of facility and in one State of the EU is fixed with the result that the expected number of those inspections is smaller than one. Therefore, in order to determine the distribution *and* timing of unannounced interim inspections a so-called State Level Approach would be required. First attempts into this direction have already been made in [1].

5. Acknowledgement

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Design Information as the basis for effective Safeguards. The case of MYRRHA.

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Abstract:

Design information (DI) is basic to effective safeguards. DI is used by the Safeguards Authorities to establish the facility safeguards approach, to determine material balance areas (MBAs) and select key measurement points (KMPs), and to establish the essential equipment list (EEL). Knowledge of the technical processes involved is mandatory.
In accordance with INFCIRC/153 para 48, the DI is verified at the facility, throughout the facility lifecycle, to allow considerations given to changes or modifications in operating conditions, developments in safeguards technology or experience in the application of verification procedures.
MYRRHA, an Accelerator Driven System (ADS) under development at SCK•CEN, aims to serve as a basis for the European experimental ADS to provide protons and neutrons for various R&D applications, to power a subcritical Pb/Bi Gen IV type reactor, and to implement Partitioning and Transmutation of Minor Actinides in high level waste.
The MYRRHA DI is evaluated against the "Safeguards by Design" principle, as put forward by the IAEA, in a recent technical workshop on the topic.

Keywords: Nuclear Safeguards; Design Information; Accelerator Driven Systems; Minor Actinides Transmutation

1. Introduction

In October 2008, the IAEA organised a Technical Workshop on “Facility Design and Plant Operation Features that Facilitate the Implementation of IAEA Safeguards” [1]. Although in accordance with INFCIRC/153 paragraph 48 [5], the DI is verified at the facility, throughout the facility lifecycle, one observes, generally speaking, an overall lack of awareness of designers and developers with safeguards requirements. The safeguards thinking should start with the pre-construction phase, and safeguards measures could be built in, in close collaboration with the Safeguards Authorities. This would be advantageous and cost-beneficial to all parties involved.

In this paper we collect the design information and investigate the safeguards modalities and possibilities for the MYRRHA project under study at SCK•CEN. At the same time, we search answers for optimizing the process of Design Information Examination (DIE) and its appropriate timing, and to remove the criticism related to the safeguards awareness in the design phase [1].

The DI is presented in Section 2 and is extracted from the technical documentation made available by the MYRRHA development team [2, 3, 4], and is based on the last information available. The new development MYRRHA will be confronted with the findings of the IAEA Technical Workshop [1], and an inspection strategy is evaluated against the IAEA Safeguards Criteria [7].
Section 3 deals with the Safeguards Approach for the MYRRHA development, linked to its actual stage in the development. A conclusion is given in section 4.

The task was part of the Belgian Support Programme to the IAEA, under reference BEL C 1746.

2. Technical description of MYRRHA – Design Information

Since 1998 SCK•CEN, in partnership with many European research laboratories, is designing a multipurpose Accelerator Driven System (ADS) for R&D applications –MYRRHA– and is conducting an associated R&D support programme. MYRRHA aims to serve as a basis for the European experimental ADS to provide protons and neutrons for various R&D applications. It consists of a linac proton accelerator delivering a “350 MeV at 5 mA” to “600 MeV at 2mA” proton beam to a windowless liquid Pb-Bi spallation target that in turn couples to a Pb-Bi cooled, subcritical fast core of 50 MW_{th}.

2.1. Design Objectives

MYRRHA is designed as a multi-purpose facility [2] to support research programmes on fission and fusion reactor structural materials and nuclear fuel for ADS, for critical reactors of present generation targeting higher burn up limits or for next generation reactors and for the production of radioisotopes for medical purposes. MYRRHA will mainly be a major contribution to demonstrate on one hand the ADS concept at a reasonable power level and on the other hand the technological feasibility of transmutation of Minor Actinides (MA) and Long-Lived Fission Products (LLFP) arising from highly radioactive waste (from reprocessing). It will also help the development of the Pb-alloys technology needed for the LFR (Lead Fast Reactor) Generation IV concept.

The MYRRHA concept is based on the coupling of a proton accelerator with a liquid Pb-Bi windowless spallation target, surrounded by a Pb-Bi cooled sub-critical neutron multiplying medium in a pool type configuration with a standing vessel. Details on the different components are given in the following paragraphs.

2.2. The Accelerator

The accelerator is a LINAC that provides the high energy protons that create the neutrons in the spallation target, needed to feed the subcritical core. The proton beam characteristics of “350 MeV at 5 mA” (and in a later version “600 MeV at 2 mA”) allow to reach high flux levels ($\Phi_{tot} > 5\times10^{15}$ n/cm²s) and a fast neutron flux of 1×10^{15} n/cm²s (E>0.75 MeV) at the Minor Actinides irradiation position under the geometrical and spatial restrictions of the sub-critical core and the spallation source. The time structure of the beam is pulsed operation with beam interruptions of 200 μs every second. A beam stability of 1% in terms of energy, 2% in terms of intensity and 10% in terms of size is foreseen.

2.3. Spallation target

The performances of an ADS in terms of flux and power levels are dictated by the spallation source strength, which is proportional to the proton beam current at a particular energy and by the sub-criticality level of the core. The sub-criticality level of 0.95 has been considered as an appropriate level for a first type medium-scale ADS. The maximum reactivity injection due to incidental conditions in the MYRRHA systems have been evaluated to about 3% that would lead to a maximum k_{eff} of 0.98 that leaves still 2% margin to the criticality.

The spallation circuit connects directly to the beam line and ultimately to the accelerator vacuum. It contains a mechanical impeller pump and a Liquid Metal/ Liquid Metal heat exchanger to the pool coolant (cold end). For regulation of the position of the free surface on which the proton beam impinges (whereby this defines the vacuum boundary of the spallation target), it comprises an auxiliary Magneto Hydro Dynamic pump. Further on, it contains services for the establishment of proper vacuum and corrosion limiting conditions.

The spallation target circuit is fully immersed in the reactor pool and interlinked with the core but its liquid metal content is separated from the core coolant. This is a consequence of the windowless design presently favoured in order to use low energy protons on a very compact target at high beam power density in order not to loose on core performance.

2.4. Fuel design

Mixed plutonium-uranium oxide fuel (MOX) with a maximum content of 30 %wt. reactor grade Pu has been chosen as the driver fuel in the pre-design of the MYRRHA sub-critical core. The reactor fuel pins have an active length of 600 mm arranged in hexagonal assemblies of 85.5 mm flat-to-flat including the fuel assembly canister thickness. MOX fuel was selected as the candidate for its better neutronic properties in a fast neutron spectrum than uranium dioxide. However, the compatibility with lead alloy coolant has been demonstrated only for uranium dioxide in a limited range of temperature and exposure to irradiation. The maximum attainable burnup is 100 MWd/kg-HM, depending on the mechanical and physical constraints on the fuel, but a demonstrated value for the sodium based fast reactor. This value can be obtained by the operation regime described in 2.7.

This fuel choice should still be checked against the non-proliferation requirements imposed to new test reactors by the RERTR (Reduced Enrichment fuel for Research Testing Reactors) programme launched by DOE US in 1996 and supported, in general, by the EU, Russian Federation and IAEA [8].

2.5. MYRRHA neutron characteristics

The neutronic design of the MYRRHA sub-critical core is based on MOX classical fast reactor fuel technology. The fuel assembly design had to be adapted to the Pb-Bi coolant characteristics especially for its higher density as compared to Na. The core configuration has been conceived with typical fast reactor hexagonal fuel assembly with a modified cell pitch to answer the Pb-Bi Eutectic (LBE) constraints. The fuel assembly has 91 fuel pins per assembly allowing a larger flexibility in the core configuration design. The reactivity value in the MYRRHA core of such a fuel assembly is ranging between ~ 0.45 to 1.6%.

The active core height is 600 mm and the maximum core radius is 1000 mm with 99 hexagonal positions, as shown in figure 1. Not all the positions are filled with fuel assemblies.

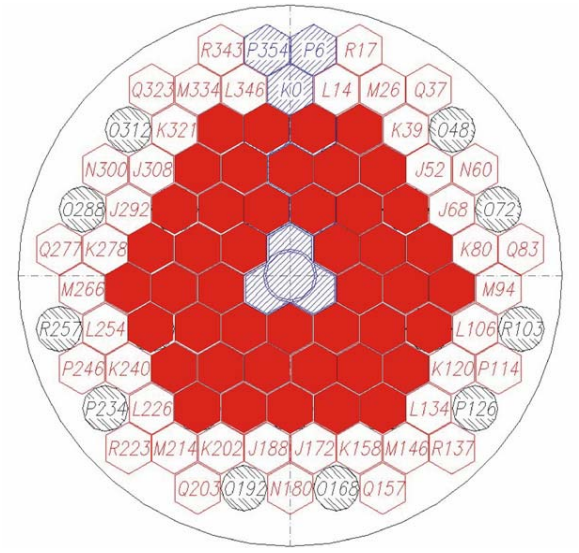


Figure 1: Core configuration for MYRRHA.
The three central hexagons are left free for housing the spallation module.

The positions could contain moderating material to create thermal neutron flux trap with Φ_{th} ~ 10^{15} n/cm²s or used as fast spectrum irradiation positions. A typical MYRRHA configuration with k_{eff} of 0.95 can be achieved by using 45 to 50 fuel assemblies. There are 17 core positions accessible through the reactor lid capable of housing experimental devices equipped with their own operating conditions control supplied by services above the reactor lid. All the other position can house either fuel assemblies or non-on-line serviced experimental rigs.

Calculations were made to compare the MYRRHA neutronics characteristics obtained with a 350 MeV proton beam to those of a 600 MeV one. The expected performances [4] in terms of fast and thermal fluxes, linear power in the core and total power in MYRRHA are given in Table 1.

Neutronics Parameters	Units	MYRRHA draft 2	MYRRHA XT-ADS
Proton beam energy	MeV	350	600
Accelerator current	mA	5	2
Proton beam heating	MW	1.43	0.74
Source neutron yield per incident proton	n/p	6.0	15.6
neutron source Intensity	10^{17} n/s	1.9	1.9
Initial fuel mixture	MOX	$(U\text{-}Pu)O_2$	$(U\text{-}Pu)O_2$
Initial (HM) fuel mass (m_{fuel})	kg	514	514
Initial Pu fuel mass	kg	131	131
Initial Pu-enrichment (Pu/HM)	wt%	30	30
k_{eff}		0.955	0.955
Thermal Power with E_f = 210 MeV/fission (P_{th})	MW	51.75	51.27
Specific power	kW/kgHM	101	100
Peak linear Power (hottest pin)	W/cm	352	324
(-5 cm ≤ Z ≤ 5 cm)-averaged neutron fluxes			
Φ_{total} fast core near hottest pin	10^{15} n/cm^{2}s	4.11	3.86
$\Phi_{>1\ MeV}$ fast core near hottest pin		0.79	0.65
$\Phi_{>0.75\ MeV}$ fast core near hottest pin		1.02	0.86

Table 1: MYRRHA facility performances

The total power of MYRRHA is about 50 MW_{th} with acceptable values of the radial and axial power form factors, 1.1 and 1.3 respectively in the hottest fuel assembly. The fast (E > 0.75 MeV) and total neutron fluxes in the core and in the neighbourhood of the spallation target reach a value of about 1.0 10^{15} and 5.0×10^{15} n/cm²s respectively.

2.6. Mechanical design of MYRRHA primary system

The design is based on two basic requirements: first, as an irradiation device, MYRRHA must have the capability to host irradiation rigs inside and outside the core, implying easy and frequent fuel reshuffling and flexible core access, and second, as a transmutation tool it requires a high fast neutron flux, implying a compact core with a high thermal loading on the fuel. This makes liquid metal cooling almost mandatory. LBE is chosen for its low melting point and high boiling point and because it doesn't react violently with air or water.

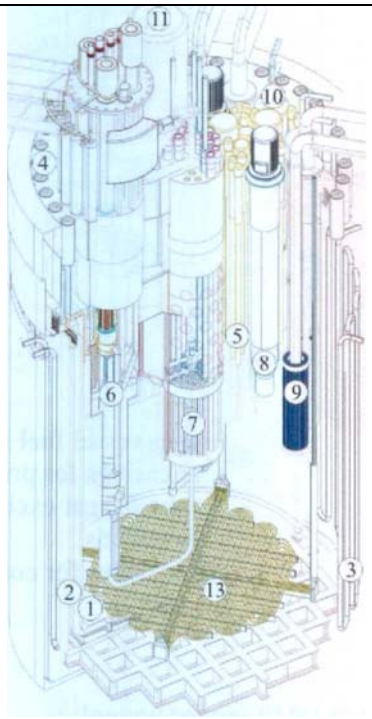
An overview and the general characteristics are given in Figure 2 and Table 2.

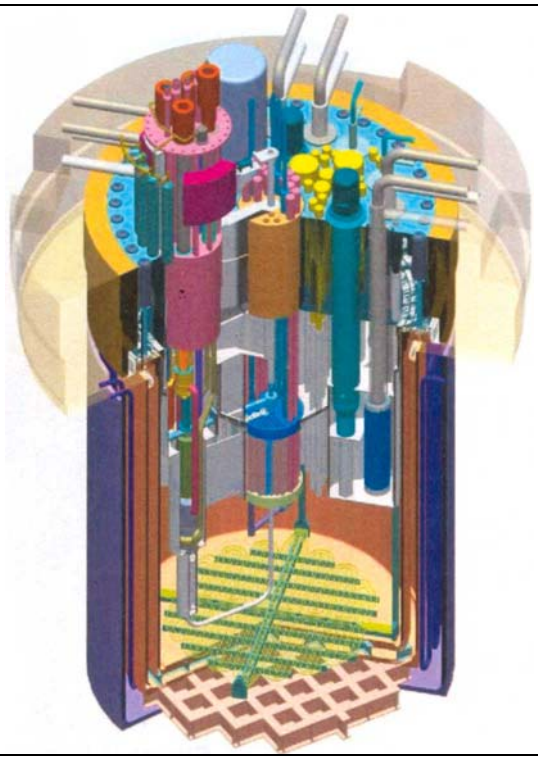
Core diameter	1000 mm
Core height	1800 mm
Fuel length	600 mm
Vessel inner diameter	4400 mm
Vessel total height (cover not included)	7000 mm
Vessel internal volume	~100 m^3
LBE volume	~65 m^3
Vessel cover thickness	~2 m
Nominal core power	50 MWth
Primary coolant	LBE
Core inlet temperature	200 °C
Core average outlet temperature	337 °C
Coolant maximum velocity in the core	2.0 m/s
Primary coolant flow rate (nominal)	2500 kg/s
Secondary coolant	Water

Table 2: General characteristics of MYRRHA

LBE serves as primary coolant for both the spallation target and the core. The primary cooling system uses water as secondary coolant. An emergency cooling system with emergency heat exchangers is foreseen. The diaphragm is a large conical component separating the lower part (high pressure, low temperature) from the upper part (low pressure, high temperature) of the vessel inner volume. To keep

the concrete inner wall at moderate temperature, a cooling screen is placed in the space between the outer vessel and the concrete pit. This screen consists mainly of vertical tubes in which a coolant circulates at low temperature (water or air 40°C).





1. inner vessel
2. guard vessel
3. cooling tubes
4. cover
5. diaphragm
6. spallation loop
7. sub-critical core
8. primary pumps
9. primary heat exchangers
10. emergency heat exchangers
11. in-vessel fuel transfer machine
12. in-vessel fuel storage
13. coolant conditioning system

Figure 2: MYRRHA 3-D Vertical View Main components

The device has a double-wall pool containment vessel with an inner diameter of 4.4 m and a height close to 7 m. It is surrounded by a biological shield to limit the activation of the surrounding soil as the MYRRHA sub-critical reactor will be installed in an underground pit. This shield will be closed above the vessel lid by forming an α-compatible hot cell and handling area for all services to the machine.

2.7. System operation, inspection & maintenance

The MYRRHA operation fuel cycle will be determined by the k_{eff} drop as a function of the core burn-up. Based on the SCK•CEN experience with the BR2 Material Test Reactor and on common rules in applications at other irradiation facilities (thermal and fast spectrum ones), MYRRHA will be operated on basis of a 3 months calendar cycle followed by a 1 month shut down for regular preventive maintenance and experiments loading/unloading as well as core reshuffling. After the third cycle it is foreseen to have a longer shut down period of three months instead of the regular one month.

To guarantee the safe operation of an ADS, subcriticality must be assured at any time. Possible excess reactivity has to be compensated through different means. First of all, there will be the inherent mechanisms that introduce antireactivity like the Doppler effect, temperature feedback effects on fuel and moderator/coolant. Additional possibilities would be the use of burnable absorber (either integrated in a fuel assembly or as a separate assembly placed around the core), control rods or fuel assembly-like boxes filled with helium at low pressure (this uses the negative void coefficient of the Pb-Bi coolant to introduce antireactivity).

The permanent monitoring of the subcriticality level is of major importance for a safe operation of an ADS. Since the dynamics of a critical reactor are quite different from the dynamics of a source-driven subcritical system, the conventional control system of a critical reactor which relies on the monitoring of the flux (flux meters) and the flux derivative (period meters) will have to be adopted to the specific needs of an ADS.

As the final goal is to provide an accurate and robust on-line measurement of the reactivity during every phase of the operation no single experimental technique can accomplish this task. Since only a very few techniques can act as an on-line indicator for reactivity and they require additional information to extract the reactivity, additional measurement techniques are needed. Therefore only a combination of techniques is able to solve the task and a step-wise and in-depth approach of reactivity determination is proposed. More details on the reactivity monitoring can be found in [4].

All in-service & repair and maintenance operation on the machine primary systems and associated equipment are performed by remote handling (RH), reducing the personnel exposure. For the maintenance operations outside the vessel, a remote handling system based on the so-called “Man-In-The-Loop” principle will be used. The operator carries out a task in the RH control room under full vision by closed-cycle TV systems while the service manipulator carries out the corresponding task in the MYRRHA Hall an arbitrary distance away. For inspections and repair inside the vessel, a remotely controlled system with an ultrasonic camera to allow visualization under LBE is foreseen for inspection of all accessible areas. If a failure is detected, a second remotely controlled manipulator (the In-Vessel Repair Manipulator) will be deployed to allow the recovery or repair of the item. The use of the ultrasonic system for the "visualization" under LBE could also be used for Safeguards purposes. The development of an ad-hoc radiation detector could also be considered, in combination with the visualization system.

3. Safeguards requirements

3.1 Legal requirements

According the IAEA Safeguards Manual SMI3.1.2, design information communication with the IAEA is a fundamental request, as mentioned [7]:

“A State party to an INFCIRC/153-type safeguards agreement or to an INFCIRC/66-type safeguards agreement is obliged to provide the IAEA with information on the nuclear material subject to safeguards under the agreement and the features of facilities relevant to safeguarding such material [7, paras. 8, 42-45]. Design information for existing facilities should be provided to the IAEA during discussion of the Subsidiary Arrangements; in the case of new facilities, the information should be provided as early as possible before nuclear material is introduced into a new facility. Further, the State should provide preliminary information on any new nuclear facility as soon as the decision is taken to construct, or to authorize the construction of the facility and should provide further information on the safeguards-relevant features of facility design early in the stages of project definition, preliminary design, construction and commissioning. Design information should be provided in respect of any safeguards-relevant change in operating conditions throughout the facility life cycle.”

Belgium is a signatory of the NPT and of the Additional protocol in the European context, and so submitted to the dual verification activities of both inspection organizations. So, the Safeguards Authorities have a higher degree of flexibility in the inspection negotiations and verification activities [5].

3.2. Design Information relevant facts

The MYRRHA project is actually in the conceptual design phase, while a parallel effort is made to assure financing, on a European level, aiming at operation at an international scale. The device anyhow runs on nuclear materials, although in a configuration that is not known to the IAEA and Euratom so far.

The most significant aspects of the DI related to the Safeguard Approach are:

- the reference concept is partially installed underground; it consists of a long and relatively thin building of 100 m length, 30 m width and 40 m height, of which 30 m are already under the ground.
- the machine hall is maintained under oxygen-free atmosphere;
- the entire spallation target was designed to be compatible with the MYRRHA remote handling system. The loop can be removed from the main vessel after core unloading. This avoids criticality risk, improves safety and allows in-situ commissioning of the spallation loop. In addition, a separate sub-unit containing all active components is foreseen allowing servicing without removal of the complete spallation loop. The closed outer housing allows yearly replacement of the spallation zone (because of embrittlement) and replacement of the heat exchanger when needed.
- the operational time of the accelerator and proton energy are known;
- the proton accelerator is not exclusively intended for MYRRHA operation. It is considered a versatile tool for scientific research, and more users already expressed interest in its use. So the time that the accelerator is operational cannot be exclusively allotted to MYRRHA operation;
- the quantity of nuclear (fissile) material is kept constant for all charges, in a geometry that can only slightly vary, so that k_{eff} is kept at value 0.95;
- there is LEU, NU, DU in the installation or building, and not only MOX (30%). All fuels can be present, but the MOX fuels are stored under the liquid metal, and only accessible through the robots.
- there is free access to the installation in as far as safety rules are respected. But neither MYRRHA, nor the fuel are any longer accessible once brought under the liquid metal.
- the systems used to monitor the reactivity could be used for Safeguards purposes.
- the whole system is by itself sealed and inaccessible, but possible extra seals could be meaningful to allow the inspectorates to draw their independent conclusions.
- the time needed for charging the complete reactor is in the order of days
- an assessment of the maximum quantity of Pu that can be bred around MYRRHA, starting from depleted uranium, and depending on the available positions in the core, essentially the periphery, was done based on the findings of Binford study [9]. This study supposes MTR fuel, pool- or tank-type reactor, light-water moderator, water, Be or graphite reflector, power level higher than 25 MW_{th}. Under these hypothesis and assuming a 50 MW_{th} power for MYRRHA, the maximum quantity of Pu that can be bred is calculated to be 1.4 SQ/year. It is unclear, at this point in time, whether the Binford study is applicable to MYRRHA but this will be investigated later on.
  In the past a comprehensive study on high-power (above 25 MW_{th}) research reactors was carried out and reported in [6], and contains a reference to the BOR-60 fast breeder sodium-cooled reactor. For this type of reactor, of Russian origin and having the same thermal power as MYRRHA, the maximum Pu production was reportes to be 1.54 SQ/year, so a 10% higher value compared to the calculation according the Binford estimation.
- the fuel will be stored in the reactor, in such quantity equal to 3 full reactor cores, under LBE, so inaccessible and not directly visible in the traditional way. So, the special criteria should apply for difficult-to-access fuel items [7].
- for inspections and repair inside the vessel, a remotely controlled system with an ultrasonic camera to allow visualization under LBE is foreseen. The use of the ultrasonic system for the "visualization" under LBE could also be used for Safeguards purposes.

3.3. Safeguards Approach and Inspection strategy

The safeguards approach is a set of safeguards measures established for the implementation of safeguards in a given situation in order to meet the applicable safeguards objectives [6, para 28]. For a given facility, it is based on the following:

- determination of possible diversion strategies and pathways
- determination of potential misuse of the facility
- determination of the appropriate safeguards measures required to meet the IAEA Safeguards criteria.

The different elements are described in the following.

Possible diversion scenarios and misuse of the installation:

- Pu assembly missing
- complete or partial replacement of Pu by LEU
- Pu, DNLEU replacement by dummy
- diversion of MOX fuel, declared to be stored under liquid metal cooler
- breeding of Pu in the experimental channels
- falsification of the operation records of the linac to hide ADS operation

According the Safeguards Criteria, verification activities are applied to the following material categories:

- fresh fuel (unirradiated direct use) MOX pins and MOX assemblies
- core fuel (irradiated) MOX assemblies
- spent fuel, pins, plates, assemblies
- closed containers
- dummy fuel assemblies

The safeguards requirements derive from the IAEA Safeguards Criteria [7], partim "Research Reactors" and "Other Reactors", and are summarized hereafter. Verification activities will include:

1. physical inventory verification
2. verification of domestic and international transfers
3. verification of other inventory changes
4. confirmation of the absence of unrecorded production of direct-use material from material subject to safeguards
5. confirmation of the absence of borrowing of nuclear material
6. material balance evaluation
7. verification activities at interim inspections for timely detection
8. verification of design information
9. verification of operator's measurement system
10. confirmation of transfers

The applicable methods for independent verification are the traditional ones: gamma instruments (HM-5, MMCA, IMCA), neutron instruments, weighing. Their applicability is hampered by the non-visibility of the fuel, once stored under the liquid Pb-Bi eutectic (so, Cerenkov devices will be of no use).

MYRRHA being practically inaccessible, the special Criteria for Difficult-to-Access Fuel Items will apply, which includes (according the Safeguards Criteria):

1. For fuel items designated difficult-to-access, evaluation of the C/S system as a whole as acceptable may serve as a basis for drawing safeguards conclusions for material balance evaluation and for timely detection, provided that all the following conditions are met.
    a. The DDG-SG has approved the designation of the fuel items as difficult-to-access due to design features of the facility. Those design features are verified at appropriate frequencies to confirm that the difficult-to-access condition has not changed.
    b. The nuclear material is verified prior to its becoming difficult-to-access by item counting, item identification (where feasible) and NDA, using sampling plans that provide a high detection probability for gross and partial defects, and dual C/S is applied.
    c. The conditions specified in SMC 14, Annex 3 for dual C/S systems to achieve acceptable C/S are met.
2. When the Dual C/S system is evaluated as either Acceptable dual C/S (both C/S systems Conclusive positive) or Acceptable Single C/S (one C/S system Conclusive positive and one C/S

Inconclusive), no remeasurement is required. For any other evaluation, follow-up actions will be decided by the DDG-SG on a case-by-case basis.

Therefore, the requirements are the following:

- approval by the DDG-SG of the difficult-to-access status of MYRRHA;
- regular verification to confirm continuation of the difficult-to-access status;
- verification of fuel before it enters the closed containment (IC, ID, NDA);
- application of dual C/S measures; e.g. seals and surveillance

SMC14 Annex 4 reveals the importance of the applied dual C/S measures. It is equally important not to hamper facility operation, or to interfere with the safety regulations. Possible solutions of independent C/S systems are sealing the reactor cover, and installation of a camera over the reactor cover.

Therefore, there is no principal incompatibility between SMC14 and technical characteristic of the device, despite the fact that the inspection modalities have still challenging aspects.

Reactivity monitoring in MYRRHA is described in paragraph 2.7. Reactivity data could as well be used for safeguards purposes, provided authentication can be assured.

Fuel can be measured by existing methods and devices, relying on gamma spectrometry and thermal/fast neutron detection. Ad-hoc detectors could be developed and used with the remote handling system.

4. Conclusions and perspectives

MYRRHA was presented in a context of defining the verification modalities for safeguards purposes.
To reach this aim, a full technical description of the device, reflecting the actual status, was presented. But it has definitely to be kept in mind that the project is still evolving, and that various technical details can still be modified.

MYRRHA is a multi-purpose research facility, where main importance has been attached to flexibility in the applications. MYRRHA satisfied the technical requirements of new installations, and represents a complementary tool to thermal neutron capability in Europe. It is responding the objectives of the experimental ADS Facility in terms of demonstration and performance, and responding by design to some key issues related to the LBE ADS. The safety requirements were fulfilled as well.
It already becomes clear from an analysis of the available technical information that the MYRRHA type installation differs substantially from existing installations in quite some points (inaccessible fuel, fast neutron spectrum, non-visibility, closed environment), and that the safeguards verification of such installation has some challenging aspects.

In determining the safeguards approach for MYRRHA, one observes similarities with as well RRCAs as with breeder reactors. The fuel is inaccessible, and stored under Pb/Bi, and as such, not visible. This means that we are in a situation called "practical inaccessible", for which the IAEA Safeguards Criteria foresee special arrangement and conditions: approval of the situation by the Director Safeguards, verification that this situation is maintained, and does not change over time, and application of dual C/S, with regular verifications.
Fuel can also be verified before being transferred in that inaccessible situation.
Several operator data are available as well as operator's technical tools.

Considering the continuing development, a regular exchange of information with the safeguards inspectorates seems a necessity.

5. Acknowledgements

Thanks are due to the MYRRHA team for providing information and for useful discussions.

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SESSION 11

NON DESTRUCTIVE ASSAY – MONTE CARLO AND MODELLING

Setting up and Benchmarking an MCNP Model for the XRF Branch of the Hybrid K-Edge/K-XRF Densitometer

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Abstract:

A mathematical simulation approach based on the general purpose Monte Carlo N-Particle transport code MCNP was developed to predict the response of the XRF branch of the Hybrid K-Edge/K-XRF Densitometer (HKED). The respective MCNP models for two different versions of HKED instruments currently in use were set up and experimentally validated. The setting up of the models involved comprehensive simulations of a bremsstrahlung photon source, the examination of different particle transport models, as well as the examination of different photon attenuation and X-ray fluorescence data libraries. The computation speed was significantly increased through the extensive use of the variance reduction techniques. The models were validated through the series of benchmarking experiments performed with a representative set of uranium, plutonium and mixed U/Pu reference solutions. The models and simulation approach developed are intended for: (i) establishing a consistent mathematical calibration approach for the XRF branch of the HKED instruments, which will require minimum calibration effort and time, (ii) extending the applicability of the HKED method to non-standard samples (e.g. U/Pu mixtures with unusual element ratios) and non-standard sample matrices (e.g. HM matrices from the pyro-processing of irradiated nuclear fuel) without investing a great deal of extra calibration work, and (iii) improving the accuracy of the measurements through the modelling of special measurement effects (e.g. the secondary excitation effect, the interference with X-ray escape peaks, the inconsistent unfolding of the overlapping peaks and peak background delineation in the measured XRF spectrum), which are difficult or sometimes impossible to account for experimentally.

Keywords: uranium; plutonium; solution; hybrid densitometry; modelling; Monte Carlo

1. Introduction

The Hybrid K-Edge/K-XRF densitometer (HKED) [1] has become a highly accurate and reliable NDA method for the determination of the U and Pu concentrations in solutions. The method is meanwhile routinely used for nuclear material accountancy verification in International Safeguards in large-scale nuclear reprocessing facilities worldwide, e.g. at La Hague (France), Sellafield (UK), Rokkasho (Japan). Through the combination of the K-edge transmission (KED) and KX-ray fluorescence (XRF) techniques the method is capable of measuring concentrations of U and Pu in solutions with a combined relative uncertainty of < 1 % at concentration levels > 0.5 g/L. The XRF branch usually covers the lower concentration range (< 50 g/L), where the K-edge transmission measurements are starting to suffer from lacking measurement precision and accuracy. In practice the XRF branch is used either as a stand-alone technique for absolute concentration measurements, or as a

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complementary to the K-edge absorption densitometry for the accurate determination of U/Pu element ratios, e.g. in the reprocessing input solutions or dissolved MOX samples [2].

The typical HKED instrument includes an X-ray generator, a sample chamber with sample position(s) for KED and XRF measurements, and low-energy HPGe planar detectors in the respective branches. The X-ray generator usually operates at 140-150 kV acceleration voltage and electron beam currents of ~10-15 mA producing an intense broad energy spectrum of bremsstrahlung photons. In the KED branch, these photons are strongly collimated and filtered and passed through a sample vial of well-defined path-length. This allows the determination of the element concentrations from the amplitudes of absorption edges in a transmission spectrum. The fluoresced KX-rays are detected at a large backward angle (typically at 150^0) with respect to the incident X-ray beam [3]. This particular XRF geometry kinematically suppresses the undesirable contribution of inelastically scattered photons in the energy region of the characteristic KX-rays of actinides (90 keV $\leq E_{KX} \leq$ 120 keV).

There exist two versions of the HKED instrument (Fig. 1): In version 1 (referred to as HKED-1) the KED and/or XRF measurements are performed on a single (cylindrical) PE sample container with an inner diameter of 14 mm. This version is in use in the Safeguards On-Site Laboratories at La Hague and Rokkasho. It is equipped with a sample changer with up to 6 sample positions [4]. Version 2 of the HKED (referred to as HKED-2) utilizes two separate sample containers, namely a glass cuvette with a path-length of 20 mm (occasionally also 40 mm) for KED and a cylindrical PE vial with an inner diameter of 9 mm for XRF [3]. This version is in use at ITU and in the Safeguards On-Site Laboratory in Sellafield.

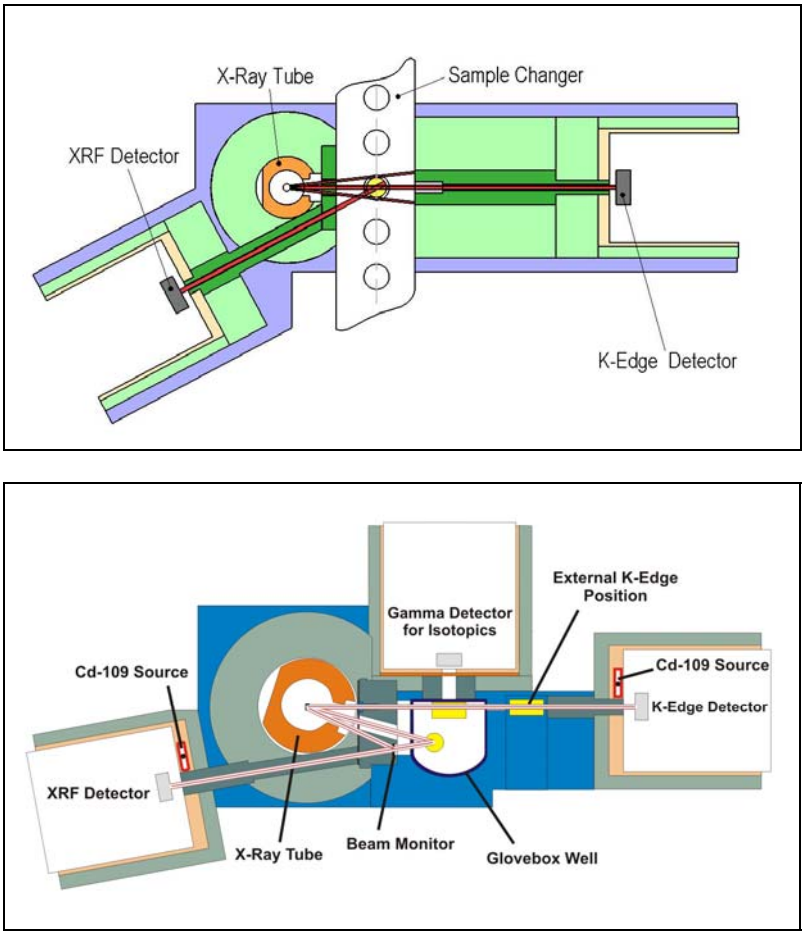


Figure 1: Schematic drawings of two existing versions of the HKED instrument. Top: Configuration using a single sample vial for KED and XRF (HKED-1). Bottom: Configuration with two separate sample vials for KED and XRF (HKED-2). Note: the gamma detector shown in the diagram is optional and does not belong to the standard HKED configuration.

In this paper we describe the Monte Carlo based models for the XRF branch in both versions of the HKED instrument. The main objective of the models is to reliably calculate the XRF response in dependence on the sample density and elemental composition, with the ultimate goal being the implementation of a reliable, versatile and time-efficient mathematical calibration of the respective XRF measurements made with the HKED.

2. Monte-Carlo modelling of the XRF measurements

2.1. General approach

The XRF measurement process consists of the following major steps: (i) the emission of a photon from the excitation source and its transport through the section of the sample volume intersected by the X-ray beam, (ii) the interaction of the photon with a target element (U or Pu) in the analysed sample with subsequent emission of the characteristic fluoresced KX-rays, and (iii) the detection of the fluoresced KX-rays in the XRF detector. With the Monte Carlo modelling these steps are simulated in a probabilistic way using a detailed mathematical model of the measurement setup and a wide scope of the physical models for transporting the photons through the instrument geometry.

For the implementation of the Monte Carlo based models for the XRF branch of the HKED instruments a general purpose Monte Carlo N-Particle transport code MCNP [5] was chosen. A specific approach employed in this work was to separate the detection part from the modelling of the rest of the HKED instruments. With this approach the flux of the characteristic X-rays at the input window of the detector cap is calculated in the first step. This flux is then translated in the following step into the XRF detector response by applying the respective detector response functions or full energy peak efficiency calibration (calculated or experimental).

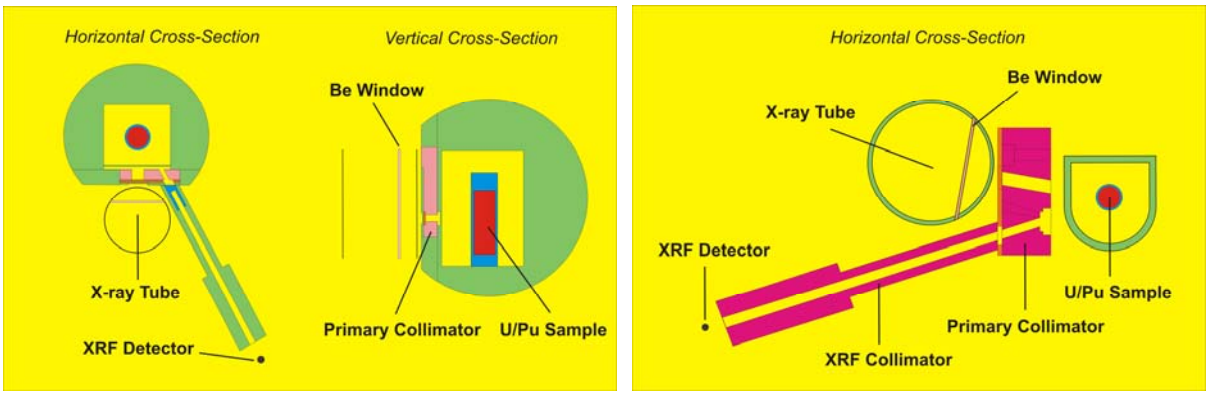
The Monte Carlo models presented here are intended for the prediction of the relative quantities rather than the absolute values of the XRF response. Such quantities are the X-ray peak intensity ratios usually used for the actinide content ratio determination or the shapes of the XRF calibration curves used for the absolute concentration measurements. The following sub-sections describe the most important components of the models developed.

2.2. Geometry models of the HKED instruments

The established MCNP models were set up to exactly reproduce the measurement geometry and conditions according to the design specifications of the respective HKED instruments. The cross-sectional views of the models for the HKED-1 and HKED-2 instruments are shown in Fig. 2. For the configuration of HKED-2 using two separate sample vials only the XRF branch was reproduced in the respective MCNP model, assuming the absence of a significant interference with the respective KED branch.

The XRF response of the HKED instruments is known to be fairly sensitive to the geometry and dimensions of the sample container, which therefore required a particular attention during modelling. The container models could be however significantly simplified owing to the ideal cylindrical shapes of the routinely used sample containers within the solid angles of the collimated incident photon beams in both versions of the HKED instrument. In the adopted models the sample containers are therefore represented by two enclosed cylinders (see Fig. 2) with an inner diameter of 14.2 in the case of HKED-1, and of 9 mm in the case of HKED-2.

A special attention was paid to the accurate reproduction of the position, shape and material properties of all absorbing layers along the paths of the incident and fluoresced X-ray beams. For the path of the primary X-ray beam from the X-ray tube these layers include the 1 mm Be window of the X-ray tube, the primary beam filter of 1 mm Cd and the 0.5 mm thick stainless steel window of the instrument chamber, and a short plug of PEEK material inserted at the front of the XRF collimator (for HKED-1 only).



(a) (b)

Figure 2: The MCNP models of the HKED-1 (a) and HKED-2 (b) instruments.

2.3. Bremsstrahlung photon source model

The HKED employs an external metal-ceramic X-ray tube as a source of the highly intense bremsstrahlung photon radiation. The continuum energy photons are produced in a tungsten target, which is bombarded by a focussed 15 mA beam of 150 keV electrons at an incidence angle of 23^0 [3]. The bremsstrahlung photons leaving the tube at $\approx 90^0$ with respect to the incident electron beam are directed into the rectangular-shaped primary collimator aperture of the HKED instrument. A fast and accurate simulation approach for the bremsstrahlung source was needed to achieve the required relative accuracy and performance of the calculations. To this end extensive MCNP simulations of the X-ray production from the tube were performed.

The bremsstrahlung photon spectra were simulated in different points of the primary collimator aperture, particularly, in the centre and two pairs of extreme points in the horizontal and vertical directions. The relative deviations between the spectral distributions in the extreme points and the spectrum in the centre are shown in Fig. 3. The largest deviations (up to 4% at the endpoint energy) are observed in the spectra in the top and bottom points of the collimator, which is due to the different magnitude of the self-attenuation effect in the W target in these extreme cases. As a result, the spectrum at the bottom is more enriched with the high-energy and less enriched with low-energy photons than the spectrum at the top. The change of the spectrum shape along the horizontal direction of the collimator aperture is not so significant, as shown in Fig. 3. It shows < 1% uniform decrease in the photon intensity over the spectrum range above 70 keV in the extreme points.

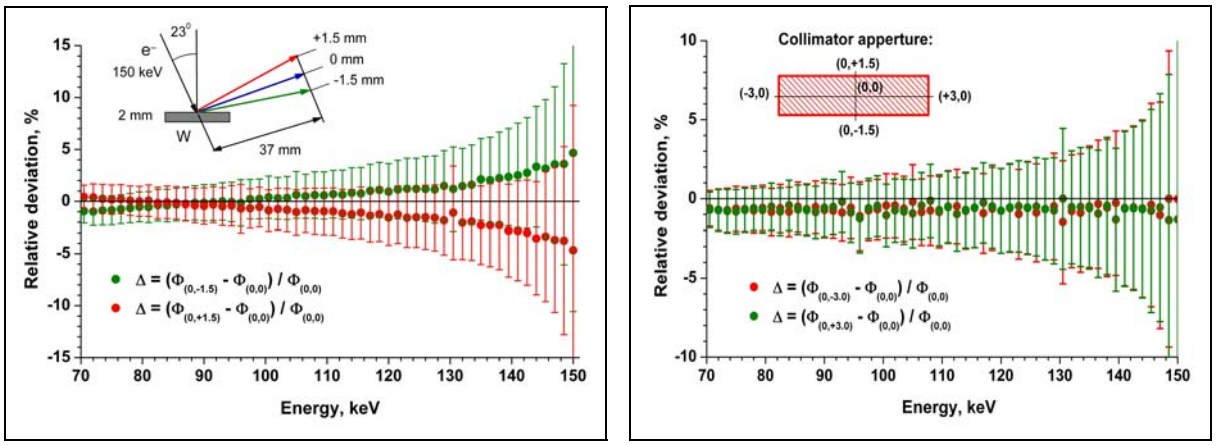


Figure 3: The relative deviations of the bremsstrahlung spectra in the extreme points of the primary collimator aperture in the vertical (to the left) and horizontal (to the right) directions from the bremsstrahlung spectrum in the centre of the collimator aperture.

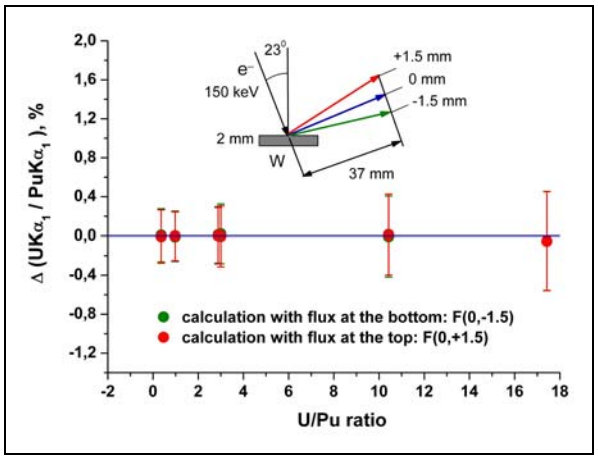


Figure 4: The relative deviation of the $UK\alpha_1/PuK\alpha_1$ ratios evaluated using the bremsstrahlung spectra in the top and bottom points of the primary collimator from the ratios evaluated using the spectrum in the centre of the collimator aperture.

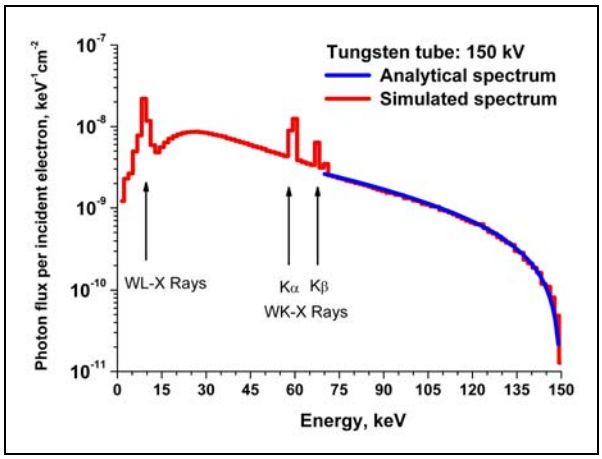


Figure 5: The comparison of the MCNP simulated bremsstrahlung spectrum with the distribution given by the semi-empirical thick-target bremsstrahlung formula [6].

The extent of the influence of the revealed spectrum variations on the XRF response was evaluated in a next step. For this purpose the intensity ratios of the characteristic U and Pu KX-rays were calculated using the bremsstrahlung spectra obtained in the previous simulations. Particularly, the spectra in the centre, top and bottom points of the collimator aperture were taken consecutively as an excitation spectrum in three separate calculation runs done for a number of concentrated U/Pu solutions with U/Pu ratio ranging from 0.3 to 17. The calculation results did not reveal any significant influence of the spectrum shape variation on the $UK\alpha_1/PuK\alpha_1$ (shown in Fig. 4) and on the $UK\beta_{13}/PuK\alpha_1$ ratios over the tested range of the heavy metal (HM) concentrations. The conclusion drawn was that the variations of the bremsstrahlung photon spectrum over the collimator aperture do not influence to any noticeable extent the XRF response of the considered HKED instruments, thus, with a good approximation, these variations can be neglected.

In a final step the simulated spectrum shapes were compared with the predictions of a semi-empirical analytical formula describing the energy distribution of the bremsstrahlung photons emitted in the given direction from a thick W target with energies above 69.53 keV (the K-absorption edge of tungsten) [6]. The agreement between the simulated and analytically calculated spectra was found to be excellent (see Fig. 5). Therefore the analytical formula was further used for tabulating the bremsstrahlung photon spectrum in the adopted MCNP models of the HKED instruments. To increase the efficiency of the calculations, this spectrum is truncated such that the sampled photon energies start from the lowest HM's K-absorption edge in the analysed sample (e.g. from the 115.61 keV for U/Pu mixtures), thus saving computer time by disregarding the transport of the useless low-energy photons. A further major enhancement of the simulation efficiency is achieved by directing the source photons towards the primary collimator (directional source biasing), and by the assumption of an identical shape of the bremsstrahlung spectrum over the collimator aperture. The latter simplification, which avoids complications in the simulations and a loss of simulation efficiency, becomes justified according to the above-described investigations.

2.4. Optimisation of computation performance

The evaluations made for the typical HKED instrument geometry show that, depending on the target element concentration, total HM content and density of the solution under assay, $10^7 \div 10^{10}$ source photons emitted towards the primary collimator with energy greater than the K-absorption edge of the respective target element (U or Pu) are needed to get one fluoresced KX-ray at the input window of the XRF detector. This represents a real challenge for the Monte Carlo based simulations in terms of the computational efforts involved, especially for the simulations aimed at achieving highly accurate and precise results. In this particular situation the efficient simulation scheme can be only implemented if based on the essentially non-analog transport models, i.e. through the extensive use of variance reduction techniques. In addition to the source energy and geometry biasing described above, the following variance reduction options offered by the MCNP code are employed in the

developed models of the HKED instruments [7, 8]:

- Geometry splitting / Russian roulette: more photons are forwarded in the preferable direction (i.e. towards the analysed sample) by generating new particles according to the assigned "importances" of the geometry cells;

- Forced collisions: the photons that reach the sample volume are forced to collide, thus, ensuring that each photon entering the analysed solution undergoes interaction;

- Point detector (next-event-estimator): the contribution of photons from every collision in the sample volume to the flux at the detector location point is counted as if these photons were emitted towards the detector. This ensures that each fluoresced KX-ray generated in the sample contributes to the final result and does not get lost;

- Selective scoring: only the events in the sample sub-volume that is "visible" to the XRF detector are scored, thus, the computing time is saved by disregarding events from unimportant regions of the sample volume and other geometry cells.

To control for eventual biases, which might result from the application of the variance reduction techniques, the simulation results were checked against the results of the respective analog calculations and available reference experimental data each time after the next optimisation method had been implemented.

With the fully optimised models a statistical precision of the XRF response from a concentrated U/Pu solution of better than 0.5 % can be achieved within a 30 min calculation run using a standard single processor PC (e.g. AMD X2 3800 2 GHz, Windows XP) and MCNPX Version 2.6. The testing of other versions of the MCNP code, such as MCNP4c, MCNP-CP, MCNP5, did not show a significant change in the computation efficiency or any noticeable difference in the calculation results. The performance of the simulations carried out with the use of an advanced computer system (IBM System x3550 Intel Xeon X5270 3.5 GHz, Linux SUSE) improved proportionally to the CPU frequency.

2.5. Photon transport models

The simulation accuracy strongly depends upon the underlying physical models and approximations. Therefore the detailed physics option offering the most accurate treatment of all basic photon interactions was used in the simulations. The process of coherent scattering, however, was excluded on purpose from the modelling in accordance with a general recommendation for the simulation schemes using point detectors. This is because of the highly peaked-forward angular distribution of the coherent scattering that can produce large and sudden contributions to the point detectors, making it practically difficult to control the statistical precision of the calculation results [7].

The drawback of this approximation however is a possible biasing of the calculation results. It may arise from disregarding the actual angular deflection of the coherently scattered photons that starts to become noticeable at photon energies below ~100 keV. For example, the calculations showed that with the excluded coherent scattering the predicted $UK\alpha_1/PuK\alpha_1$ peak intensity ratios can be overestimated by about 0.5%. This seems to present a sort of unavoidable bias in the presently accepted modelling approach.

The quality of the photoatomic data is also of great importance for achieving the required accuracy of the simulations. The evaluation of the available options revealed discrepancies in the energies and intensities of the X-rays between the most recent (MCPLIB04) and earlier (MCPLIB, MCPLIB02 and MCPLIB03) versions of the MCNP photoatomic data library [9]. Particularly, the energies of the $UK\alpha_1$ and $PuK\alpha_1$ lines in the MCPLIB04 turned out to be shifted by about 0.5 keV to higher values with respect to the data in the earlier versions.

Numerous tests showed however that despite this discrepancy the relative XRF response (e.g. the peak intensity ratios and calibration curve shapes) is insensitive to a particular set of the photoatomic data within the achieved calculation precision. For the purpose of this study the MCPLIB03, providing the KX-ray energies and intensities in agreement with the generally accepted values, was adopted as the source of the photoatomic data.

The simplified treatment of the Kβ-series X-rays is another drawback of the MCPLIB photoatomic data library. Instead of five Kβ-lines the MCPLIB considers two $K\beta'_1$ and $K\beta'_2$ "lines", which are weighted averages over $K\beta_1$-$K\beta_3$-$K\beta_5$ and $K\beta_2$-$K\beta_4$, respectively [9]. This turned out to be inappropriate for obtaining a sufficient agreement between the calculated and experimental peak intensity ratios involving Kβ-lines.

3. Experimental validation of the XRF models

3.1. U/Pu weight ratio measurements

In this analysis mode the U/Pu element ratio is usually obtained from the $UK\alpha_1/PuK\alpha_1$ peak intensity ratio evaluated from the measured XRF spectrum. This ratio can be calculated using the developed modelling approach, thus the direct comparison with the experiment is possible.

In the validation experiment a set of nitric acid solutions of MOX samples was used. The U and Pu element concentrations in the solutions were determined by IDMS. The ^{241}Am content was quantified by HRGS. The solution densities were determined experimentally with the Anton Paar densitometer according to routine procedures providing 0.05% measurement uncertainty at the 95% confidence level. The solution properties are listed in Table 1.

The measurements were performed with the HKED-2 type instrument available in ITU. Each sample was measured five times for 5000 s to determine the influence of an eventual instrument instability on the measurement uncertainty. The stability of the acceleration voltage of the X-ray tube was controlled via the end-point energy of the bremsstrahlung continuum in the transmission spectrum accumulated in parallel. The accumulated spectra were analysed with the standard Canberra HKED software.

No.	Density, g/cm^3	Molarity	U, mg/g	Pu, mg/g	U/Pu weight ratio	^{241}Am, mg/g	HM, mg/g
1	1.62304	11.75	139.32	13.38	10.41	0.127	152.83
2	1.58172	12.19	122.34	7.02	17.42	0.056	129.42
3	1.58025	12.31	92.76	30.86	3.01	0.235	123.85
4	1.55218	12.30	106.01	7.09	14.95	0.057	113.16
5	1.61744	12.00	135.82	10.94	12.42	0.056	146.85
6	1.59056	12.35	124.85	7.60	16.43	0.096	132.52
7	1.55270	12.30	103.37	8.56	12.08	0.073	112.00
8	1.57173	12.30	112.77	9.14	12.34	0.075	121.99
9	1.57786	12.12	119.08	8.85	13.46	0.010	128.03

Table 1: Properties of the MOX solutions used in the validation experiment.

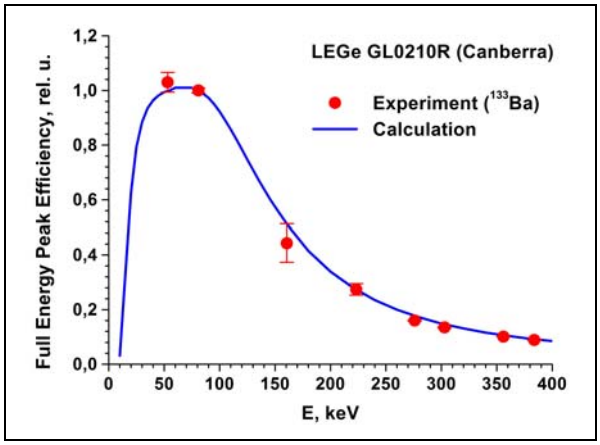


Figure 6: The detection efficiency of the HKED-2 XRF detector (LEGe, 200 mm^2 × 10 mm, Canberra) used for the $UK\alpha_1/PuK\alpha_1$ flux-to-peak intensity ratio conversion.

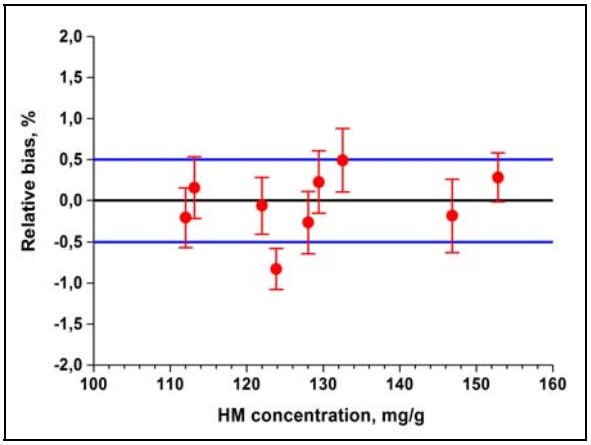


Figure 7: The relative differences between the calculated and experimental $UK\alpha_1/PuK\alpha_1$ peak intensity ratios for a set of reference MOX solutions.

The theoretical estimates of the $UK\alpha_1/PuK\alpha_1$ peak intensity ratio represent the ratio of the $UK\alpha_1$ and $PuK\alpha_1$ X-ray fluxes calculated with the developed MCNP model of the instrument. This ratio was corrected for the difference in the detection efficiencies of the $UK\alpha_1$ and $PuK\alpha_1$ photons. The full energy peak efficiency of the XRF detector (LEGe, 200 $mm^2 \times 10$ mm, Canberra) for the given counting geometry was evaluated in a separate Monte Carlo calculation and afterwards validated experimentally with the use of a ^{133}Ba reference gamma-source (see results presented in Fig. 6).

Fig. 7 provides a plot of the relative differences between the calculated and experimental results for the $UK\alpha_1/PuK\alpha_1$ peak intensity ratio. The data demonstrate for the full set of the investigated samples very good agreement (within 0.5%) between the simulation results and the experimental values.

3.2. Analysis of low U and Pu concentrations in HM matrices

Application of the Monte Carlo modelling is absolutely essential when actinide concentrations in non-standard matrices (e.g. unusual molarity, presence of other HMs etc.) are to be determined from stand-along XRF measurements. This situation will exist, for example, for measurement samples originating from the pyrochemical processing of spent nuclear fuel. The major matrices in those samples will be represented by the eutectic salt, zirconium, cadmium or bismuth, i.e. matrices completely different from ones normally encountered in the conventional PUREX process. Another peculiarity of these samples is the rather low concentrations of actinides, typically only a few weight percent, whereas the content of the mentioned matrix elements is expected to be significantly higher. For this kind of XRF measurements the Monte Carlo modelling is capable of providing the matrix corrections for the characteristic KX-ray peaks of actinides in the measured XRF spectrum.

The applicability and adequacy of the developed MCNP models of the HKED instrument to this kind of measurements was studied with the use of experimental data. These data were obtained with a set of reference solutions containing low concentrations of U (5 g/L) and Pu (2.4 g/L) and different amounts of Bi ranging from 0 to ~200 g/L. The samples were measured in the HKED-2 type instrument available in ITU. The quantities evaluated from the experimental spectra were the count rates in the characteristic X-ray peaks normalized per unit concentration of an actinide. As seen from Fig. 8, these quantities are gradually decreasing with the increase of the HM concentration. This is also illustrated by the decrease of the actinide peak intensities in the respective XRF spectra shown in Fig. 9.

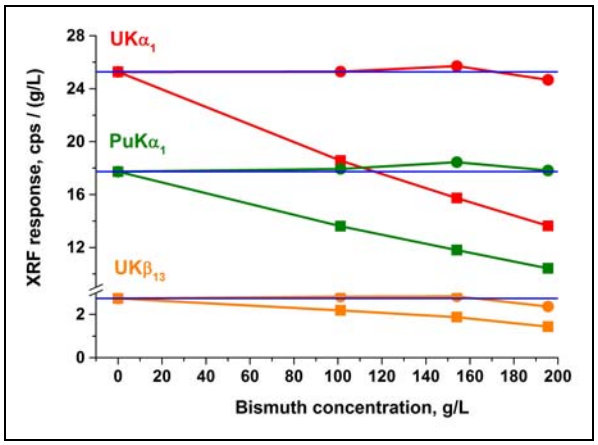


Figure 8: The XRF actinide responses from low concentration U/Pu solutions as function of Bi content. Circles represent data points corrected for the matrix effect, squares represent uncorrected data.

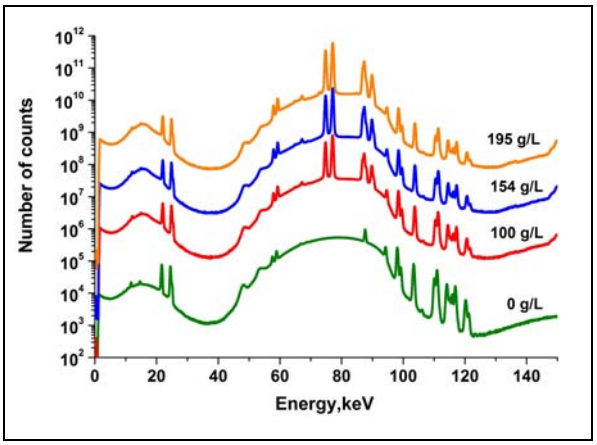


Figure 9: The XRF spectra acquired from low concentration U/Pu solutions (5 g/L U + 2.4 g/L Pu) containing increasing amounts of Bi (concentrations of Bi are indicated on the diagram).

The severe impact of the matrix effect becomes evident from the uncorrected data in Fig. 8. To correct for this effect, matrix correction factors according to the sample composition were calculated using the developed MCNP model of the HKED-2 instrument. The correction factors were evaluated as the ratio

of the KX-ray photon flux at the input window of the detector cap calculated for a pure U/Pu solution to the fluxes calculated for the solutions with given Bi concentrations. The corrected data points in Fig.8 demonstrate the independence of the XRF response on the matrix element concentration. The obtained results prove the applicability of the developed MCNP model to the analysis of low-concentration actinides in HM matrices.

3.3. Absolute actinide concentration measurements

The absolute U and Pu concentration measurements in the low concentration range (< 50 g/L) are generally made with the XRF branch of the HKED instrument in a "stand-alone" mode. The accuracy requirements for these measurements are therefore very high. Considerable efforts are usually needed to establish a relevant calibration of the instrument down to the low concentration limit of ~0.5 g/L. A practical limitation of such a calibration, however, accrues from the fact that the pertinent calibration is only valid for samples with properties (e.g. density, actinide ratios, matrix) similar to those of the calibration solutions. The modelling of the XRF response can therefore be of great benefit in this kind of measurements, because it allows to greatly reduce the calibration efforts and to extend the applicability limits of the method.

The capability of the developed HKED models to accurately reproduce the shapes of the calibration curves of the XRF branch of the instrument was tested. The experimental data for this testing were taken from the XRF calibration measurements recently performed with an HKED-1 type instrument. The calibration involved five reference solutions for both U and Pu, each measured five times for 5000 sec. The accumulated spectra were evaluated with the Canberra HKED spectrum processing software yielding values of the XRF response in terms of the count rate in $K\alpha_1$ X-ray peak of an actinide per unit concentration. The obtained experimental data points for U and Pu are shown in Figs. 10 and 12, respectively.

The calculated responses were obtained by normalizing the $K\alpha_1$ X-ray fluxes (simulated at the input window of the XRF detector end cap) to the respective actinide concentrations. The final normalization was done such that the calculated and experimental responses matched at the lower concentration limit of 0.5 g/L. The calculated and experimental calibration curves compared in Figs. 10 and 12 exhibit a good overall agreement. Figs. 11 and 13 show the relative deviation of the calculated data points from the experimental data. The results suggest an agreement of 0.5% or better between the calculated and measured XRF response in dependence on the respective actinide element concentration. The same level of agreement between calculated and measured data is also observed for the simulation results obtained with the coherent scattering (CS) included (see Figs. 11 and 13). This finding can be taken as an indication for the equality of the simulation results obtained with and without coherent scattering, at least for the given application. In our adopted simulation approach of the HKED XRF measurements the coherent scattering is not included (see pertinent discussion in section 2.5).

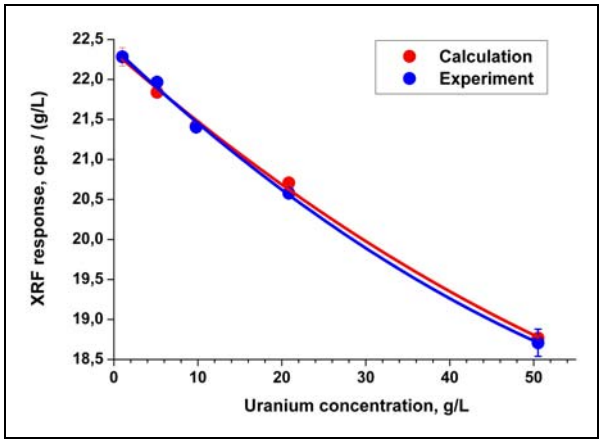


Figure 10: Experimental and simulated XRF calibration curves for the low U concentration measurements with a HKED-1 instrument.

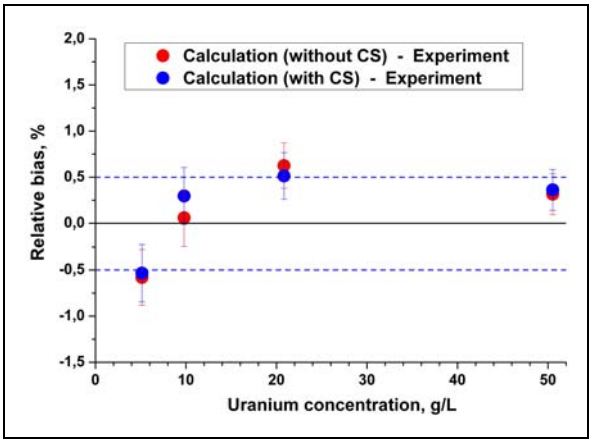


Figure 11: Percentage difference between the calculated and measured XRF calibrations for low U concentration measurements with a HKED-1 instrument.

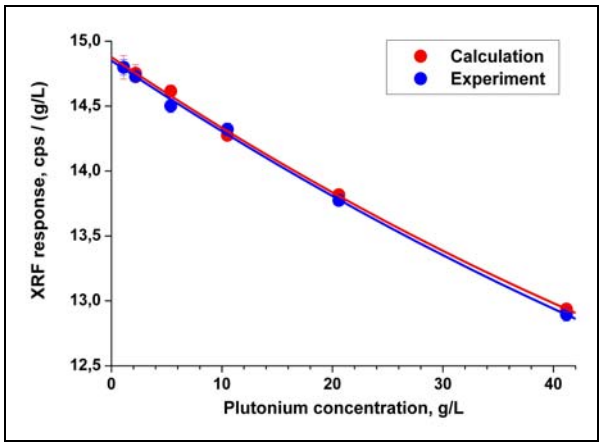


Figure 12: Experimental and simulated XRF calibration curves for the low Pu concentration measurements with a HKED-1 instrument.

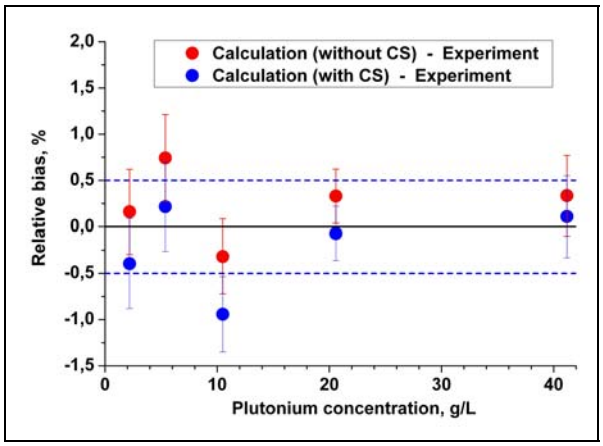


Figure 13: Percentage difference between the calculated and measured XRF calibrations for low Pu concentration measurements with a HKED-1 instrument.

4. Conclusions

The Monte Carlo based mathematical simulation approach with respective MCNP models for the XRF branch of two different versions of HKED instruments currently in use was set up and validated. The achieved accuracy (< 0.5%) and speed (30-40 min for the statistical precision better than 0.5%) of the simulations provide a firm basis for the further practical implementation of a consistent and time-efficient mathematical calibration approach of the XRF measurements in the following applications:

- the U/Pu element ratio measurement in the reprocessing input solutions and dissolved MOX product samples, where the ratio can be determined in principle solely based on the mathematical calibration of the instrument;
- the absolute U and Pu measurements in the lower concentration range (< 50 g/L), where in practice a single calibration measurement can be already sufficient to establish, together with the modelling results, a full scale calibration for the absolute XRF measurements;
- the measurement of U and Pu concentrations in non-standard matrices (e.g. unusual molarity, presence of other actinides or HMs), where modelling is capable of providing accurate matrix corrections for the yields of characteristic KX-rays of actinides from the assayed solutions.

The application of the developed models can also be of great usefulness for the development and testing of XRF spectrum processing algorithms and software. Here the special measurement effects (e.g. the secondary excitation effect, the interference with X-ray escape peaks, the inconsistent unfolding of the overlapping peaks and peak background delineation in the measured XRF spectrum) representing the potential sources of the XRF measurement biases can be easily modelled and examined.

5. Future work

The further developments of the presented modelling approach towards the implementation of a full mathematical calibration of the XRF measurements will include:

- the introduction of a complete set of Kβ-series X-rays into the MCNP modelling that seems to be crucial for obtaining an agreement with the experiment for X-ray peak intensity ratios involving Kβ-lines. To achieve this, the photoatomic data library MCPLIB has to be appropriately modified;
- the implementation of the full XRF spectrum modelling that can be a useful option for testing XRF

spectrum processing algorithms and development of measurement techniques for special samples and sample matrices;

- the creation of a database of the XRF responses of the HKED instruments on the grids of U and Pu concentrations and sample densities. This is intended for a quick evaluation of the correction factors for samples, whose properties are different from those' used in the instrument calibration.

Acknowledgements

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Results of phase III and IV of the ESARDA Multiplicity Benchmark

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Abstract:

In 2003 the ESARDA NDA working group launched a benchmark exercise in order to compare the different algorithms and codes used in the simulation of neutron multiplicity counters. The results of the 1st and 2nd phase of the ESARDA Multiplicity Benchmark, based on synthetic cases, have been published in the ESARDA Bulletin number 34. Notwithstanding the satisfactory conclusion that all the algorithms developed by the different participants in the first two phases and used to analyse the pulse trains have proven to be satisfactory, the working group felt that an extension to real experimental cases would have added a supplementary value to the exercise that brought to the organisation of phases III and IV. This paper summarises the outcomes of the benchmark, whose full report will soon be made available on the ESARDA Bulletin.

Keywords: NDA, neutron counting, neutron multiplicity, Monte Carlo

1. Introduction

In 2003 the ESARDA NDA working group has launched a benchmark in order to compare the different algorithms and codes used in the simulation of neutron multiplicity counters. In order to derive the maximum of information and at the same time to allow a large participation, the working group decided to split the exercise in two parts with two participation levels: a full simulation exercise where participants were asked to compute the count rates starting from the basic technical specifications and/or a partial exercise involving the processing of the pulse trains produced by a single laboratory. The results of participants performing the entire exercise allowed making a comparison among the different Monte Carlo codes for the simulation of neutron multiplicity counters. The results of the partial exercise help to test the available algorithms for pulse train analysis and to derive some important information about the models applied for dead-time correction. The results of the 1st and 2nd phase of the ESARDA Multiplicity Benchmark have been published in a special issue of the ESARDA Bulletin [1] and a summary presented and published in the proceedings of the 47th INMM Annual Meeting [2].

All cases run in the first two phases of the benchmark were totally theoretical. So the conclusions derived had to be considered as a relative behaviour of the different models, techniques and codes. Notwithstanding the satisfactory conclusion that all the algorithms developed by the different participants in the first two phases and used to analyse the pulse trains have proven to be satisfactory, the working group felt that an extension to real experimental cases would have added a supplementary value to the exercise.

Therefore it was decided to use an experimental campaign performed for testing LIST mode acquisition systems [3] for a continuation of the 2003 benchmark. Again it contained a first step (3rd phase) devoted to full Monte Carlo simulation and a second one (4th phase) for the inter-comparison of software for analysis of LIST mode files. For this benchmark we selected the 6 measurements performed with the IRSN set-up (AWCC with MEDAS card). The LIST mode acquisitions (for each case 10 repeated measurements of 100 s each) have been stored in binary pulse train files and

distributed to the participants who have tested their software computing Singles, Doubles and Triples rates and associated uncertainties. These were compared to the reference: the S, D and T rates obtained with an analogue-electronic acquisition (AMSR). The results of this second benchmark are currently under publication on the ESARDA Bulletin. We will summarise here the most important outcomes from the 4th phase, the most relevant one to the topic of this paper, for more details we refer to the benchmark final report (foreseen to be published soon on the ESARDA Bulletin).

2. Results of phase III

The scope of this part of the exercise is to have a comparison of the different codes available for the complete simulation of a neutron multiplicity counter. Six laboratories provided results for the full exercise. The laboratories are more or less the same who participated to the phase 1 and they used the same codes: MCNPX by LANL and IRSN, MCNP-PTA by JRC, MCNP-PoliMi by Chalmers and University of Michigan and MCNP+AM technique by IPPE. A new participant, IRSN, provided two sets of results: one with MCNPX with direct calculation of moments, the second by generating pulse trains with MCNPX and then processing the files with the post-processor TRIDEN, used also for phase 4. Methodological details have been already described in [1] and will not be repeated here.

a) Zero dead-time

Table 1 shows the comparison of the simulation results in an ideal case of zero dead-time. The quoted uncertainties are purely statistical at 1-sigma level. In this case the calculated values can be compared with the theoretical value computed using the point model [4].

It is worthwhile to note that comparison to the point model is not trivial, because it requires the knowledge of parameters like the efficiency, the leakage multiplication and the gate utilisation factors. The Pu sources are confounded by (alpha,n) neutrons with a different energy distribution and a finite extent which may violate the strict assumptions of the point model, but more importantly the items were not all measured in the same position so there will be a shift in efficiency from the centre to the floor. Some variants of the point model may imply a simple exponential die-away and the AWCC is not truly ideal in that sense. The “reference” values reported in table 1 have been computed applying the point model equations with some simplifications and approximations. The values for efficiency and multiplication were derived from the Monte Carlo calculations; this automatically accounts for variation of efficiency within the cavity, size and shape of the sample, different energy of neutrons from (alpha,n) and spontaneous fission. The doubles gate fraction was computed assuming a single exponential with an approximate die-away time of 50 μs and the triples gate fraction was assumed to be the square of the doubles gate fraction. The moments of the induced fission multiplicity distributions were taken from fast (1 MeV) neutron fission, not from thermal fission as often used.

The results of the simulations at zero dead-time show an excellent agreement among the different participants, with standard deviations within a few percent in most of the cases. It is true that all the methods have a common model for the simulation of neutron transport based on MCNP, but the methods differ on the treatment of time correlations and in any case we always expect some effects linked to the human factor (the way in which the user models the system). Taking all this into account the agreement among the results is satisfying.

Even though there is no clear evidence of strong systematic errors, some clear trends are visible. For instance all the results based on MCNPX (IRSN and LANL) tend to be consistently lower than those based on MCNP-Polimi (Chalmers); MCNP-PTA tends to overestimate Triples, whereas IPPE method tends to underestimate them.

Moreover the agreement between simulations, theoretical expectations and measurements are also good. This confirms the applicability of the point model in the cases represented in this exercise.

b) Dead-time Effects

The previous data cannot be directly compared with measurements, since measured data are affected by dead-time effects. So we have two possibilities, either we correct the measured data in order to derive zero dead-time values or the dead-time effects should be included in the simulation. Dead-time

corrected experimental values have been also included in table 1 and can be compared with the zero dead-time simulations there. In this section we have considered the second option.

MCNP-PTA allows direct modelling of dead-time for each component of the electronic chain (amplifiers and OR-chain or mixer), MCNPX can produce a pulse train file that can be post-processed by a simulation program that includes dead-time effects (in case of IRSN the TRIDEN software uses a global system dead-time). Both codes apply a paralysable dead-time model. JRC used a dead-time component of 1 μs per each of the 6 amplifiers and 20 ns for the OR-chain; this corresponds to a system dead-time of 1000/6+20 = 187 ns, consistent with the one used by IRSN (170 ns). Therefore IRSN and JRC provided as well a set of results that can be directly compared with measured values. This is reported in table 2.

When comparing simulations with measurements, we notice a less close agreement. This can only be marginally attributed to the uncertainty introduced by dead-time effects, or the way how the dead-time is modelled. There is certainly some unresolved inconsistency between the model and the reality. This is especially true for the low count rate cases where the dead-time correction is negligible. This is confirmed by the comparison of the dead-time corrections (obtained by dividing the results of table 1 by those of table 2) shown in table 3.

By consequence we have to attribute the origin of the discrepancies to modelling and, to a less extent, to the nuclear data used by the two codes. Indeed we have to keep in mind that the PERLA standards are certified with a very high precision in terms of mass and isotopic composition, but much less in terms of their geometrical properties. Container size is of course known, but there is a large uncertainty on the powder density of cases 4, 5 and 6 that is reflected on an uncertainty of the filling height and therefore on the actual sample dimensions. This affects quite strongly the multiplication and therefore introduces a systematic error that increases with the order of the moments. Especially in case 6 the density of MOX powder is not known at all (explaining the strong discrepancies in this case), whereas the density of PuO_2 (assumed to be 2.6 g/cm^3) has been obtained using some gamma scanning of the containers that allowed us to derive the powder filling height with reasonable accuracy. A similar consideration applies to case 3 where the sample thickness is not certified.

3. Results of phase IV

The LIST mode files processed from the MEDAS card acquisitions have been distributed to the participants, who were requested to compute the Singles, Doubles and Triples counting rates for all the pulse trains. For each case 1000-second acquisitions were performed, more precisely ten independent acquisitions of 100 seconds. The participants produced the S, D and T count rates (average on the ten short acquisitions) together with their absolute uncertainties. Additionally they were requested to provide indicative processing times of the pulse trains together with the PC characteristics.

All the results are reported in tabular form in table 4. Generally the agreement among the different processing codes is extremely good: negligible deviations in Singles (less than 0.1%), agreement within 0.1%-0.4% in Doubles. Nevertheless dispersion up to 4% in Triples is visible, indicating that the way to compute them is not totally homogeneous.
The values can also be compared to the measured S, D, T with a multiplicity shift register. Indeed it is one of the scopes of the exercise to assess the capability of LIST mode acquisition to correctly collect the measured data in view of a possible alternate technology in data acquisition for neutron counting applications. Indeed the results show that data acquired with the time-stamping data acquisition card and processed with all the tested codes agree with the multiplicity shift register data within the statistical uncertainties. We should bear in mind that the shift register measurements and the LIST mode measurements were done with the same experimental setup, but sequentially in time. This means that they do not refer exactly to the very same pulse train, but to two sequential pulse trains acquired in identical conditions. So we can only conclude that they coincide within the statistical uncertainty and no systematic deviations have been revealed.
One outstanding feature of these results is the quoted absolute error. The values appear to vary by an order of magnitude from group to group. This is an important issue that again should be studied by a more detailed comparison of the calculation methods and even definitions of uncertainty used by the

different groups. The results could be compared to the values from the shift register electronics and theoretical values.

A further point of interest is the processing time required which varies by more than an order of magnitude. This will be partly due to the computer processing power available but there may also be tips and tricks that could benefit the safeguards community, if, as we expect, there is widespread future use of list mode data.

Overall, the results of this exercise show that all participants are capable of good performance for practical purposes. However, comparison of the methods used by the different groups should allow the establishment of more robust analysis techniques with more reliable error estimates.

4. Conclusions

The results presented here lead to a number of interesting and important conclusions. There are two separate topics. The first is the simulation of measurements using Monte Carlo. In this area the results of the different participants are very similar. However differences do remain. This is in spite of the fact that the basic geometric model was the same in all cases. A close comparison of the input/output files used by the participants will reveal the sources of these differences. The effect of input parameters such as fill height or nuclear data is available in these results and could lead to useful information on the accuracy of simulations for Monte Carlo users.

The second part of the work involved the analysis of pulse trains. In this area also the results from the different participants are very similar. The difference between the results is small for most practical purposes. However when one considers that each team was starting from identical pulse trains it seems that further detailed comparison of the algorithms used would be warranted. One free parameter in the analysis is the length of the long delay and another difference is how the physical end of the data is treated. Otherwise the results would be expected to be truly identical. A study of the different data treatment algorithms could be used to establish a reference standard for the data analysis.

One outstanding feature of these results is the quoted absolute error. The values appear to vary by an order of magnitude from group to group. This is an important issue that again should be studied by a more detailed comparison of the calculation methods and even definitions of uncertainty used by the different groups. The results could be compared to the values from the shift register electronics and theoretical values. We just underline that from a theoretical point of view the statistical uncertainty of a measurement should not depend on the fact that the acquisition is done using a shift register or LIST mode.

Overall, the results of this exercise show that all participants are capable of good performance for practical purposes. However, comparison of the methods used by the different groups should allow the establishment of more robust analysis techniques with more reliable error estimates.

References

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Table 1 – Results from (zero dead-time) simulations and comparison with theoretical (point model) values

	Counting time	Singles rate	S abs. unc.	Doubles rate	D abs. unc.	Triples rate	T abs. unc.
Case 1: Cf low intensity							
Point model		***1170***		***380.28***		***68.84***	
Experimental (DT corrected)		***1211***		***382.64***		***67.66***	
Chalmers Univ.	1000	1175	3.85	386.56	4.25	71.66	3.25
IPPE	1000	1147	2.12	362.00	1.34	61.01	0.43
JRC	52000	1167	0.21	376.61	0.13	73.70	0.88
IRSN (MCNPX direct)		1167	0.47	374.68	0.37	67.34	0.19
IRSN (MCNPX + TRIDEN)	1000	1165	2.20	374.29	1.37	67.69	0.91
Univ. Michigan		1160		376.91		73.64	
LANL		1160	0.46	372.48	0.45	66.85	0.21
Relative standard deviation		0.01		0.02		0.07	
Case 2: Cf high intensity							
Point model		***153837***		***50011***		***9053***	
Experimental (DT corrected)		***153768***		***48545***		***8347***	
Chalmers Univ.	1000	154692	23.88	51032	238.47	9432	585.63
IPPE	1000	150770	17.83	47602	82.24	8022	219.28
JRC	402	153070	25.73	48880	123.23	8760	363.82
IRSN (MCNPX direct)		153318	61.33	49240	49.24	8850	24.78
IRSN (MCNPX + TRIDEN)	1000	153282	11.80	49318	49.07	8434	72.57
Univ. Michigan		153572		49786		8656	
LANL		151826	60.73	48738	58.49	8747	27.99
Relative standard deviation		0.01		0.02		0.05	
Case 3: Pu metal							
Point model		***724***		***142.38***		***16.72***	
Experimental (DT corrected)		***721***		***129.25***		***14.29***	
Chalmers Univ.	1000	713	3.06	139.48	1.59	16.45	0.89
IPPE	1000	716	0.91	138.65	0.85	15.35	0.22
JRC	52000	722	0.17	142.17	0.74	18.31	0.41
IRSN (MCNPX direct)		701	0.14	130.46	0.09	14.24	0.03
IRSN (MCNPX + TRIDEN)	1000	700	1.51	129.97	0.51	13.93	0.23
Univ. Michigan		708		136.13		16.89	
LANL		693	0.03	130.64	0.01	14.76	0.01
Relative standard deviation		0.01		0.04		0.10	

Case 4: Pu oxide small mass							
Point model		***7297***		***904.87***		***107.36***	
Experimental (DT corrected)		***7328***		***919.14***		***113.83***	
Chalmers Univ.	1000	6534	5.87	866.51	12.48	109.03	7.85
IPPE	1000	7317	2.83	944.07	4.10	113.96	3.08
JRC	6378	7282	1.40	945.02	1.39	126.93	1.14
IRSN (MCNPX direct)		7031	6.24	892.64	1.38	110.14	0.62
IRSN (MCNPX + TRIDEN)	1000	7072	3.42	912.41	3.65	112.86	2.22
Univ. Michigan		7196		942.05		136.16	
LANL		6962	0.92	901.04	0.61	111.58	0.26
Relative standard deviation		0.04		0.03		0.09	
Case 5: Pu oxide large mass							
Point model		***147656***		***23009***		***4224***	
Experimental (DT corrected)		***146568***		***23316***		***4595***	
Chalmers Univ.	1000	130564	33.01	20957	238.98	4093	546.71
IPPE	1000	146530	16.00	22487	71.05	3814	170.95
JRC	242	147060	31.30	23456	147.31	4802	363.75
IRSN (MCNPX direct)		146731	28.43	23748	17.39	4474	11.60
IRSN (MCNPX + TRIDEN)	1000	146818	4.38	23650	24.25	3765	36.59
Univ. Michigan		144804		22940		5137	
LANL		139974	15.10	21894	17.70	3980	8.90
Relative standard deviation		0.04		0.05		0.12	
Case 6: MOX sample							
Point model		***26157***		***3411***		***371.56***	
Experimental (DT corrected)		***27772***		***3128***		***348.26***	
Chalmers Univ.	1000	25719	10.93	3397	37.14	373.18	30.04
IPPE	1000	25504	4.00	3184	10.74	303.24	17.94
JRC	2388	26135	4.60	3414	8.53	397.14	8.44
IRSN (MCNPX direct)		24784	4.63	3109	1.87	329.09	0.72
IRSN (MCNPX + TRIDEN)	1000	24773	3.95	3124	7.59	348.58	13.53
Univ. Michigan		21552		3123		349.58	
LANL		23507	2.50	2991	2.40	336.21	0.80
Relative standard deviation		0.06		0.05		0.09	

Table 2 – Comparison of measurements and simulations with dead-time effects

	Singles rate	Doubles rate	Triples rate
Case 1: Cf low intensity			
measured	**1208.08**	**380.73**	**66.62**
MCNP-PTA	1164.60	374.73	71.96
(C-E)/E	*-3.6%*	*-1.6%*	*8.0%*
MCNP-Polimi	1157.74	375.63	72.60
(C-E)/E	*-4.2%*	*-1.3%*	*9.0%*
MCNPX + post-processor	1162.61	372.82	66.65
(C-E)/E	*-3.8%*	*-2.1%*	*0.0%*
Case 2: Cf high intensity			
measured	**149338**	**43374**	**3695**
MCNP-PTA	148660	43673	3743
(C-E)/E	*-0.5%*	*0.7%*	*1.3%*
MCNP-Polimi	149976	45438	4550
(C-E)/E	*0.4%*	*4.8%*	*23.1%*
MCNPX + post-processor	149145	44398	3734
(C-E)/E	*-0.1%*	*2.4%*	*1.0%*
Case 3: Pu metal			
measured	**720.51**	**129.09**	**14.07**
MCNP-PTA	701.41	133.32	15.07
(C-E)/E	*-2.7%*	*3.3%*	*7.1%*
MCNP-Polimi	707.63	135.83	16.65
(C-E)/E	*-1.8%*	*5.2%*	*18.4%*
MCNPX + post-processor	698.72	129.50	13.69
(C-E)/E	*-3.0%*	*0.3%*	*-2.7%*
Case 4: Pu oxide small mass			
measured	**7313.46**	**912.24**	**109.29**
MCNP-PTA	7267.50	937.93	121.09
(C-E)/E	*-0.6%*	*2.8%*	*10.8%*
MCNP-Polimi	7183.27	935.75	130.34
(C-E)/E	*-1.8%*	*2.6%*	*19.3%*
MCNPX + post-processor	7059.04	906.40	108.36
(C-E)/E	*-3.5%*	*-0.6%*	*-0.9%*
Case 5: Pu oxide large mass			
measured	**142622**	**20873**	**2519**
MCNP-PTA	143100	20998	2632
(C-E)/E	*0.3%*	*0.6%*	*4.5%*
MCNP-Polimi	141739	21156	3696
(C-E)/E	*-0.6%*	*1.4%*	*46.7%*
MCNPX + post-processor	143143	21499	2053
(C-E)/E	*0.4%*	*3.0%*	*-18.5%*
Case 6: MOX sample			
measured	**27623**	**3064**	**301.8**
MCNP-PTA	25995	3344	344.2
(C-E)/E	*-5.9%*	*9.1%*	*14.0%*
MCNP-Polimi	21471	3078	320.3
(C-E)/E	*-22.3%*	*0.5%*	*6.1%*
MCNPX + post-processor	24654	3066	303.40
(C-E)/E	*-10.7%*	*0.1%*	*0.5%*

Table 3 – Comparison of dead-time correction factors

	Singles	Doubles	Triples
Case 1: Cf low intensity			
MCNP-PTA	1.002	1.005	1.024
MCNP-Polimi	1.002	1.003	1.014
MCNPX + TRIDEN	1.002	1.004	1.016
Case 2: Cf high intensity			
MCNP-PTA	1.03	1.12	2.34
MCNP-Polimi	1.02	1.10	1.90
MCNPX + TRIDEN	1.03	1.11	2.26
Case 3: Pu metal			
MCNP-PTA	1.001	1.003	1.016
MCNP-Polimi	1.001	1.002	1.014
MCNPX + TRIDEN	1.002	1.004	1.018
Case 4: Pu oxide small mass			
MCNP-PTA	1.002	1.008	1.048
MCNP-Polimi	1.002	1.007	1.045
MCNPX + TRIDEN	1.002	1.007	1.042
Case 5: Pu oxide large mass			
MCNP-PTA	1.03	1.12	1.82
MCNP-Polimi	1.02	1.08	1.39
MCNPX + TRIDEN	1.03	1.10	1.83
Case 6: MOX			
MCNP-PTA	1.005	1.021	1.15
MCNP-Polimi	1.004	1.015	1.09
MCNPX + TRIDEN	1.005	1.019	1.15

Table 4 – Results from experimental pulse train processing and comparison with multiplicity shift register measurements

	Counting time	Singles rate	S abs. unc.	Doubles rate	D abs. unc.	Triples rate	T abs. unc.	Time
Case 1: Cf low intensity								
Measured		***1247.87***	***1.58***	***380.78***	***0.84***	***66.65***	***0.48***	
Chalmers Univ.	1053.32	1244.42	4.49	382.26	2.94	67.09	2.23	3.4
IPPE	1050.00	1244.44	1.43	381.70	0.77	66.39	0.57	1.0
CEA-DAM	1053.32	1244.34	4.49	381.88	2.43	66.72	2.99	0.5
CEA-LMN		1243.94	1.57	381.90	1.14	66.74	0.75	0.7
AREVA	1040.00	1244.30	4.37	382.07	3.87	72.52	2.74	8.2
JRC	1053.24	1244.40	4.48	381.26	2.94	66.23	2.12	2.0
JRC-2	1053.32	1244.37	4.50	381.80	2.82	66.67	1.98	2.8
IKI	954.90	1243.90	1.10	382.60	1.20	67.30	2.00	1.0
IRSN	1053.29	1244.39	1.42	381.60	0.75	66.35	1.75	5.9
CANBERRA	1053.20	1244.70	4.61	381.32	3.50	66.37	2.51	0.9
Univ. Michigan	1053.30	1244.36		381.96		72.48		
LANL	1053.25	1244.39	1.42	382.23	0.92	67.07	0.71	0.8
Rel. stand. dev.		*0.000*		*0.001*		*0.034*		
Case 2: Cf high intensity								
Measured		***149378.13***	***7.30***	***43373.58***	***33.22***	***3695.29***	***71.23***	
Chalmers Univ.	1000.39	149364.24	45.84	43522.80	178.00	3142.29	352.84	4030.0
IPPE	1000.00	149360.00	15.22	43454.00	66.91	3310.80	161.28	34.0
CEA-DAM	1000.39	149362.55	45.87	43522.68	231.62	3403.39	551.33	689.3
CEA-LMN		149364.27	16.74	43470.35	77.52	3266.27	189.99	45.3
AREVA	990.00	149364.15	43.03	43492.82	245.88	3785.99	455.87	972.1
JRC	1000.34	149364.22	45.80	43525.16	286.15	3353.66	469.35	4989.0
JRC-2	1000.38	149364.26	45.87	43513.48	251.58	3328.03	644.65	218.0
IKI	999.06	149364.00	12.00	43543.00	65.00	3334.00	646.00	17.0
IRSN	1000.38	149364.26	14.50	43470.39	64.04	3271.56	144.81	2267.9
CANBERRA	1000.30	149364.63	59.18	43532.54	244.70	3554.16	627.31	1075.0
Univ. Michigan	1000.40	149360.80		43484.06		4053.14		
LANL	1000.34	149364.21	14.49	43522.92	55.69	3145.53	112.13	113.0
Rel. stand. dev.		*0.000*		*0.001*		*0.032*		

Case 3: Pu metal								
Measured		***760.288***	***1.123***	***129.141***	***0.566***	***14.099***	***0.303***	
Chalmers Univ.	1269.51	761.16	3.34	130.84	1.68	14.16	0.68	2.5
IPPE	1270.00	761.15	1.06	130.62	0.46	13.95	0.20	0.0
CEA-DAM	1269.51	761.12	3.34	131.07	1.53	14.20	0.65	0.3
CEA-LMN		760.10	1.03	130.42	0.59	14.00	0.28	0.6
AREVA	1000.00	760.10	3.45	130.45	1.57	15.50	0.88	3.4
JRC	1269.42	761.17	3.34	130.62	1.55	14.13	0.48	2.0
JRC-2	1269.51	761.14	3.34	130.86	1.66	14.23	0.72	2.4
IKI	1108.76	760.50	0.90	130.70	0.60	14.40	0.80	1.0
IRSN	1269.48	761.15	1.06	130.68	0.44	13.98	0.19	4.3
CANBERRA	1269.40	761.33	3.09	130.63	1.52	14.16	0.62	1.1
Univ. Michigan	1269.50	761.43		130.68		15.50		
LANL	1269.41	761.16	1.06	130.84	0.53	14.16	0.22	0.8
Rel. stand. dev.		*0.001*		*0.001*		*0.009*		
Case 4: Pu oxide small mass								
Measured		***7353.24***	***4.43***	***912.29***	***3.76***	***109.32***	***2.83***	
Chalmers Univ.	1070.66	7345.31	11.26	913.10	8.62	110.78	6.96	28.3
IPPE	1070.00	7345.30	3.56	906.72	3.94	106.56	2.55	3.0
CEA-DAM	1070.66	7345.18	11.26	910.12	9.43	110.11	8.22	4.1
CEA-LMN		7345.65	3.09	906.43	3.66	107.00	2.35	2.2
AREVA	1000.00	7345.65	12.16	910.25	10.89	117.06	7.27	29.9
JRC	1070.62	7345.30	11.28	908.96	9.20	108.53	6.17	37.0
JRC-2	1070.66	7345.27	11.25	906.70	6.84	109.61	7.89	11.6
IKI	1053.90	7345.40	2.60	907.40	3.30	110.00	7.70	1.0
IRSN	1070.65	7345.27	3.56	906.65	4.12	106.49	2.66	38.3
CANBERRA	1070.60	7345.47	9.24	908.95	10.54	108.76	6.44	8.0
Univ. Michigan	1070.60	7345.64		907.41		118.75		
LANL	1070.61	7345.29	3.57	913.10	2.73	110.78	2.21	3.7
Rel. stand. dev.		*0.000*		*0.003*		*0.015*		

Case 5: Pu oxide large mass								
Measured		***142661.62***	***16.19***	***20873.04***	***40.18***	***2518.91***	***175.20***	
Chalmers Univ.	1001.80	142611.58	33.83	20940.06	255.57	2459.96	702.97	3700.0
IPPE	1000.00	142610.00	10.85	20909.00	63.13	2432.50	142.18	33.0
CEA-DAM	1001.80	142609.95	33.91	20949.26	164.21	2473.05	551.68	630.0
CEA-LMN		142611.03	14.21	20913.85	63.36	2420.11	141.24	44.1
AREVA	1000.00	142611.03	33.80	20931.81	237.37	2911.00	743.75	907.5
JRC	1001.76	142611.49	34.04	20925.37	209.17	2473.20	431.78	4536.0
JRC-2	1001.79	142611.59	33.92	20973.49	108.53	2562.83	566.41	208.0
IKI	1000.84	142611.00	12.00	20987.00	56.00	2568.00	572.00	16.0
IRSN	1001.79	142611.59	10.72	20912.59	67.72	2422.62	171.51	2096.9
CANBERRA	1001.80	142612.12	38.28	20934.08	183.00	2462.23	460.18	916.4
Univ. Michigan	1001.80	141606.60		20776.55		2872.58		
LANL	1001.75	142611.47	10.76	20939.33	80.74	2458.78	223.22	104.0
Rel. stand. dev.		*0.002*		*0.002*		*0.021*		
Case 6: MOX sample								
Measured		***27662.76***	***5.23***	***3063.97***	***15.55***	***301.84***	***19.31***	
Chalmers Univ.	1018.88	27658.27	17.35	3082.96	35.38	292.67	47.84	200.0
IPPE	1020.00	27658.00	5.50	3053.40	9.83	276.40	10.37	6.0
CEA-DAM	1018.88	27657.91	17.33	3059.32	18.40	303.23	40.18	31.3
CEA-LMN		27658.42	6.25	3053.55	11.82	274.67	11.99	7.8
AREVA	1000.00	27658.43	18.47	3073.31	35.94	310.16	44.15	118.4
JRC	1018.84	27658.23	17.33	3075.91	26.28	283.85	36.96	224.0
JRC-2	1018.87	27658.24	17.33	3071.09	22.21	294.35	38.00	40.0
IKI	912.89	27661.30	5.50	3071.90	10.60	281.50	37.40	3.0
IRSN	1018.87	27658.24	5.48	3052.45	9.93	276.79	10.75	179.2
CANBERRA	1018.80	27658.48	21.07	3075.86	35.71	283.15	35.56	52.1
Univ. Michigan	1018.90	27657.43		3055.31		315.23		
LANL	1018.84	27658.22	5.48	3082.92	11.18	292.69	15.15	14.2
Rel. stand. dev.		*0.000*		*0.004*		*0.031*		

Table 5 - Contributors to the "ESARDA Multiplicity Benchmark Exercise"

Participant	Contributors	Institution
JRC	Paolo Peerani Z. Dzbikowicz M. Marin Ferrer Luc Dechamp Pascal Dransart	European Commission, Joint Research Centre, IPSC, Ispra, Italy
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Monte Carlo Modelling of the Guinevere Detector and Fuel Assembly System for Nuclear Safeguards

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Abstract

A European based Guinevere (Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor) project conducted within FP5 "IP-Eurotrans", addresses issues concerned with the development of Advanced Driven Systems for the purpose of partitioning and transmutation for nuclear waste volume and radio toxicity reduction and aims to obtain a validated methodology for on-line reactivity monitoring.

A substantial part of this project consists in the modification of the VENUS water moderated criticality facility of the SCK/CEN (in Mol Belgium) into a fast lead core reactor facility which will be coupled to a purposely Modified 14 MEV neutron generator (GENEPI-C). The proposed new fuel assemblies, 88 of them each containing 9 Uranium fuel rods interspaced with lead blocks, are constructed with lead blocks and uranium fuel for the core and lead blocks for the reflectors. Within the Euratom treaty, this new fuel assemblies need to be safeguarded and verified. For this purpose, the European Commission Inspectors (DG-TREN) intend to use an Active Well Coincidence Counter (AWCC) which can not unfortunately be calibrated and used for verification in the classical sense due to the fact that there are no reference materials representing the new fuel magazines.

The solution adopted is a Monte Carlo based calibration and verification procedure which has now been fully investigated and found very sound when applied to many NDA neutron counting systems [1]. The AWCC has been fully modelled using the Monte Carlo code, MCNP-PTA, which simulate both the neutron transport and the coincidence electronics.

The AWCC model has been extensively validated following measurements carried out in the JRC-Ispra Performance Laboratories (PERLA) using our MTR materials. Measurements agreed within 1-3 % with our calculations. Furthermore the new Genuivere fuel magazines have fully modelled using MCNP, based on detailed engineering drawings kindly provided to us by the site operators (SCK/CEN).

This paper describes the Monte Carlo simulation methodology and models applied and gives the results of our measurements and calculations.

Keywords: Neutron counting, calibration, NDA, Monte Carlo simulation, Guinevere
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1. Introduction

The Guinevere (Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor) experiment, conducted within the European Commission Framework Programme 5 project “IP-Eurotrans”, addresses issues concerned with the development of Accelerator Driven Systems for the purpose of partitioning and transmutation for nuclear waste volume and radio toxicity reduction. It thus aims to obtain a validated methodology for on-line reactivity monitoring.

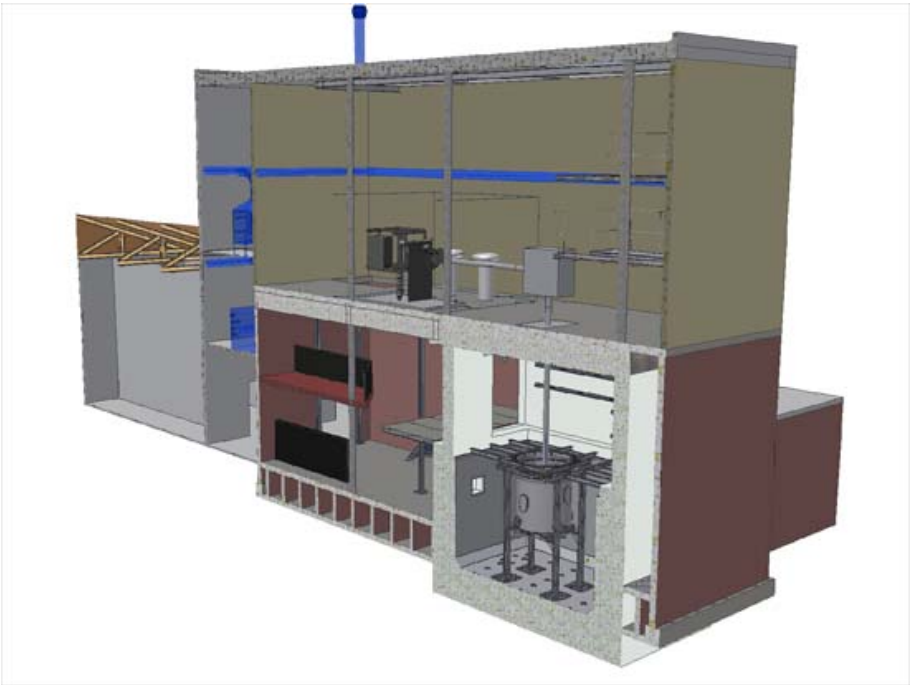

Figure 1 – VENUS facility at SCK/CEN centre in Mol

At the SCK/CEN nuclear site in MOL (Belgium) a new GENEPI-C 14-MeV neutron generator (to be built by CNRS, France) will be coupled to a modified VENUS-F zero-power fast lead reactor (figure 1). The one meter in diameter reactor core will be will host 88 fuel assemblies of about 60 cm active length totally redesigned and newly built by CEA. Each assembly contains nine uranium rods separated by lead rods as described in reference [5] and shown in figures 2 and 3.

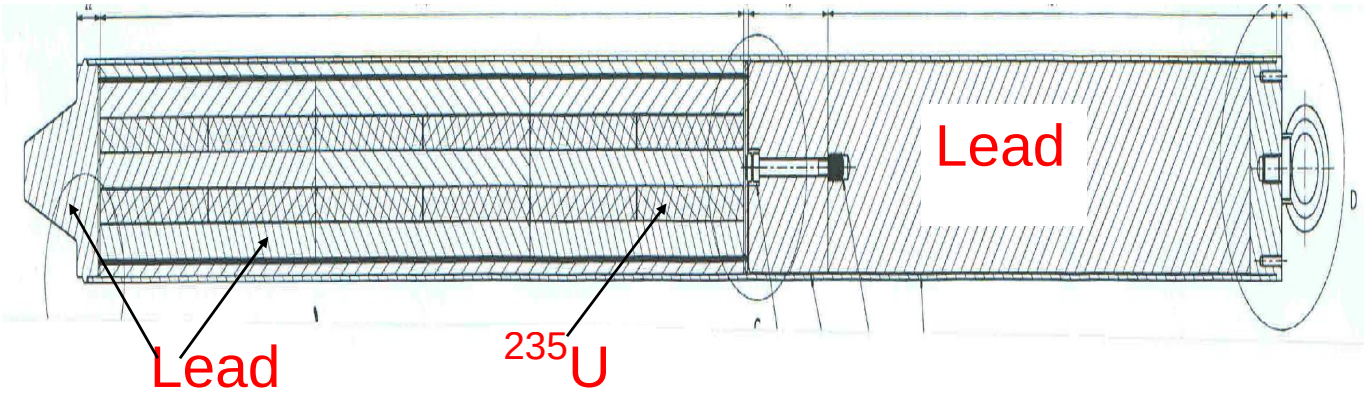

Figure 2 – Longitudinal view of a Guinevere fuel assembly

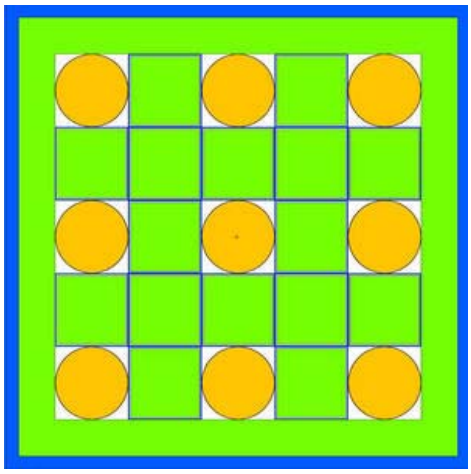

Figure 3 – Transversal view of a Guinevere fuel element:
HEU rods (yellow) and lead (green)

The fresh fuel elements will be subject to Euratom safeguards inspections to verify their fissile material content. Nuclear inspectors will use an active neutron interrogation technique. Since the Guinevere fuel is unique in its kind, there are no representative reference materials that can be used for calibrating any detection system to be employed in the traditional way for the determination of the mass of fissile material by the nuclear safeguards inspectors of Euratom. This paper discusses the solution developed by JRC to calibrate a measuring device dedicated to the verifications of Guinevere fuel.

2. Methodology

The verifications of fissile content in fresh fuel elements are usually done by active neutron interrogation. In most cases a neutron coincidence collar (NCC) is used [2]. In case of short fuel elements, as those used in some research reactors, nuclear inspectors can use an Active Well Coincidence Counter (AWCC) used in a particular configuration, called MTR-mode.

One of the major problems in active neutron counting is the fact that the instrument must be calibrated with reference fuel assemblies having the same configuration of the fuel to be verified. This is not an issue for standard traditional fuel used in reactors or in relatively common research reactors (such as MTR or TRIGA), but becomes really challenging for unique or prototypic reactors, where no reference material matching the characteristics of the fuel can be available.

JRC has developed and successfully applied a methodology based on computational calibration relying on the simulation of the measurement system with Monte Carlo codes. A similar problem had been solved in the past when calibrating an NCC to be used in the verification of the unique HEU fuel elements used in the German research reactor FRM-II [3]. In that case an NCC was used, whereas in the case of the Guinevere fuel TREN inspectors plan to use an AWCC in MTR-mode. This allows applying the same methodology, but will require performing the model validation for the different counter.

The process developed to implement computational calibration of neutron coincidence counters has been described in [1] and consists of the following steps:

1. The models of detector and fuel element are developed using MCNP-PTA.
2. HEU-MTR reference materials are measured using the same counter type (AWCC).
3. The Monte Carlo models are validated comparing measurements to calculations.
4. The response functions for the special fuel element in the AWCC are calculated.
5. On-site verification of real Guinevere elements.

We will shortly describe in this section the first step. The validation process (second and third steps) will be described in the next section and the results for Guinevere simulations will be reported in the last section. The final on-site measurements have not yet taken place at this moment.

The simulations of the response function of an AWCC loaded with 2 AmLi neutron sources and HEU fuel elements (either MTR reference materials or Guinevere fuel) have been done using the Monte Carlo code MCNP-PTA [1] developed by JRC as upgrade of the classical MCNP code developed at Los Alamos [4].

MCNP-PTA performs a full simulation of a neutron coincidence (or multiplicity) counter by coupling two main computational models: the first one describing the generation, transport and detection of neutrons in the sample+detector system (phase 1), the second (phase 2) one representing the pulse train analysis.

Phase 1 is a (nearly) standard MCNP run. Modifications with respect to the original code include the automatic generation of the inherent neutron source (including spontaneous fission and (alpha,n) reactions) based on the geometry and material definition, the detailed treatment of neutron multiplicity distributions from (induced or spontaneous) fission and some support to the rapid description of the geometry in lattices of fuel elements.

Following phase 1, information such the originating event number, detector number where the neutron is detected, time elapsed from generation to detection is saved into a file for each neutron detected. The PTA part of the code (phase 2) generates the pulse train sequence which is analysed simulating the same logic as that of the neutron analyser (e.g. a shift register). Using the instrument settings (pre-delay and gate width) and including a realistic model of the dead-time, the Totals, the Accidentals and the Reals rates are computed. MCNP-PTA has now matured and has been well validated and successfully applied to many systems.

3. Validation of the detector model

The model of the AWCC neutron counter [6] in MTR configurations has been validated in the PERLA laboratory of JRC Ispra with respect to experimental measurements done using HEU certified fuel elements of MTR type.

The MTR fuel elements available in PERLA consist in arrays of plane plates. Each plate is a three-layer sandwich of Al-fuel-Al having an active length of approximately 60 cm, a width of 7 cm and a thickness of 0.13 cm. The individual plates are removable, so any configuration between 2 plates and full elements are possible in order to build a calibration curve as a function of fuel mass.

Two different elements have been used:

- PERLA2 with up to 19 plates of 45% enriched uranium each one containing approximately 16 g of U-235
- PERLA3 with up to 23 plates of 93% enriched uranium each one containing approximately 8 g of U-235

The fuel elements with several decreasing plate arrangements have been measured in an AWCC using two AmLi neutron source for interrogation. Totals and Reals coincidence rates have been acquired by a shift-register analyser, model CANBERRA JSR-12.

All the measured configurations have been reproduced and simulated using MCNP-PTA and the results are compared in table 1. It is quite evident that there is a quite good agreement among calculations and measurements where count rates are predicted always with accuracy better than 3%.

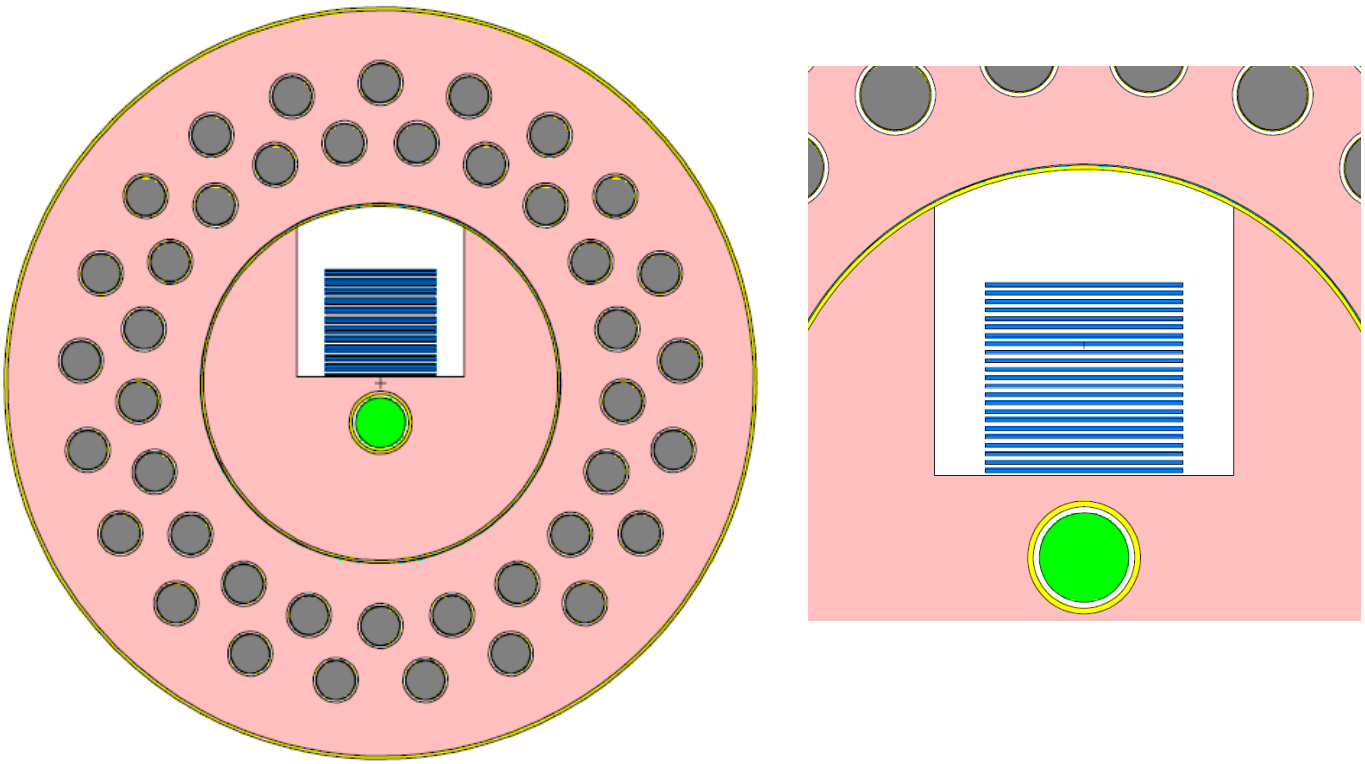

Figure 4 – Cross view of the Monte Carlo model of the AWCC
(polyethylene moderator in pink and He-3 detectors in gray)
with MTR fuel element plates (blue) and AmLi source (green)

MTR fuel	Nr. plates	Enrichm.	U-235 (g)	Measurements			MCNP-PTA calculations				C/E	
				Totals	Reals	dR	Intensity	Totals	Reals	Net R	Totals	Reals
none	0	0	0	6800	-1	4	73500	6575	1		0.97	
perla3	23	93.11%	179.60	7833	235	3	73500	7539	230	229	0.96	0.97
perla3	14	93.11%	109.34	7427	149	3	73500	7238	155	154	0.97	1.03
perla2	19	45.03%	310.29	8228	280	4	73500	8013	276	275	0.97	0.98
perla2	17	45.03%	277.62	8118	256	5	73500	7896	251	250	0.97	0.98
perla2	15	45.03%	244.98	7988	223	4	73500	7777	228	227	0.97	1.02
perla2	10	45.03%	163.15	7654	163	3	73500	7467	159	158	0.98	0.97
perla2	5	45.03%	81.53	7258	90	2	73500	7089	88	87	0.98	0.97

Table 1 – Comparison calculations versus measurements with AWCC and MTR fuel elements

4. Simulation of the Guinevere fuel and measurement device

The Monte Carlo model of the AWCC with Guinevere fuel element located into the cavity is shown in figure 5. The measurement has been simulated with MCNP-PTA assuming the use of two AmLi sources with the same intensity of those used in the validation campaign (total intensity 73500 n/s). The calculations for the full fuel magazine naturally located within the AWCC cavity resulted:

Reals = 761

Totals = 7948

R+A = 4805

The expected rates for the measurement device used on-site will have to be renormalized to the intensity of the sources used there. To this purpose the results of the normalisation measurements (AmLi sources in place, but no sample in the cavity) will be used. The on-site measured Totals (Tn) will have to be compared to the rate measured in PERLA (6575 counts per second) and the values reported above will have to be corrected with by a factor:

F = Tn / 6575.

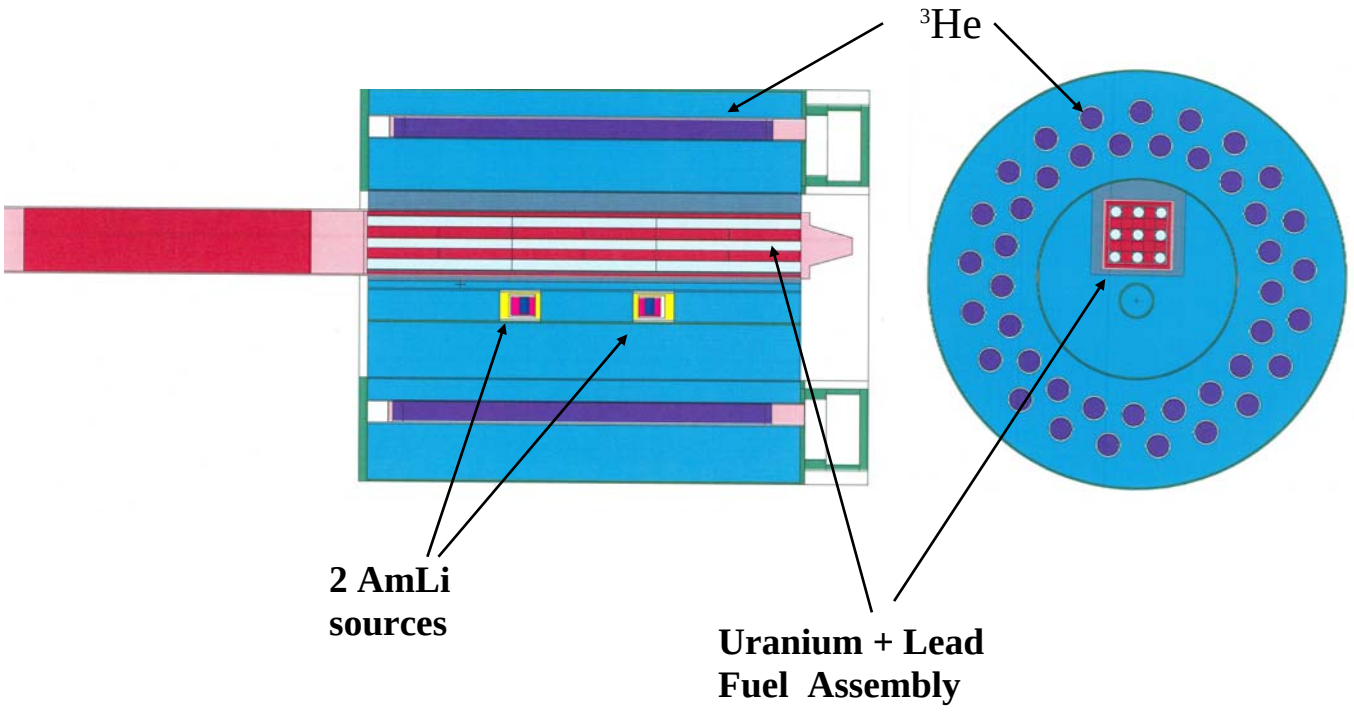

Figure 5 – Monte Carlo model of the AWCC with Guinevere fuel element

5. Conclusions

Monte Carlo simulation of neutron counters has become a mature technique finding more and more practical applications in nuclear safeguards. The methodology outlined in this paper can be considered an agreed standard for the development of computational calibration of NDA instruments.

MCNP-PTA is a fully validated simulation tool for neutron coincidence and multiplicity counters. It has been extensively applied with success in many cases where traditional calibration with certified reference material was impossible.

The application of the technique to the special case of the Guinevere fuel has been performed as a natural extension of the already consolidated procedure. The system can be considered calibrated and ready for use to Euratom inspectors.

The next and last step will be the on-site verification measurements at Mol of the fresh fuel assemblies already shipped to the reactor site from the fabrication plant.

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Unfolding sample parameters from neutron and gamma multiplicities

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Abstract:

Expressions for neutron and gamma factorial moments have been known in the literature. For neutrons, these served as the basis of constructing analytic expressions for the detection rates of singles, doubles and triples, which can be used to unfold sample parameters from the measured multiplicity rates. Here we extend also the gamma factorial moments into detection rates of multiplets, and suggest the combined use of both the individual and joint neutron and gamma multiplicities and the corresponding detection rates. Counting up to third order, there are nine auto- and cross factorial moments, which are all given here explicitly.

Adding the gamma counting to the neutrons introduces new unknowns, related to gamma generation, leakage, and detection. Despite of having more unknowns, the total number of measurable moments exceeds the number of unknowns. On the other hand, the structure of the additional equations is substantially more complicated than that of the neutron moments, hence their analytical inversion is not possible.

We suggest therefore to invert the non-linear system of over-determined equations by using artificial neural networks (ANN), which can handle both the non-linearity and the redundances in the measured quantities in an effective and accurate way. The use of ANNs is demonstrated with good results on the unfolding of neutron multiplicity rates for the sample fission rate, the leakage multiplication and the α *ratio.*

Keywords: safeguards; neutron and gamma multiplicities; joint moments; material accounting and control

1. Introduction

Multiplicity detection rates, based on higher order factorial moments of the neutron counts from an unknown sample, can be used to determine sample parameters [1–3]. The factorial moments here refer to those of the total number of neutrons generated in the sample by one *initial source event* (spontaneous fission or (α,n) reaction). Due to internal multiplication through induced fission, the probability distribution of the total number of generated neutrons will deviate from that by the initial source event (mostly spontaneous fission), the deviation being a function of the sample mass (via the first collision probability of the initial neutrons). This property is transferred to the measured multiplicity rates, i.e. the singles, doubles and triples, and this is corroborated by the fact that in the latter the sample fission rate occurs explicitly. This gives a possibility to determine the sample mass.

Measurement of the first three multiplicity rates enables the recovery of three unknowns, which are usually taken as the sample leakage multiplication $\mathbf{M}$ (related to the first collision probability p), the ratio α of the intensity of single neutron production via (α, n) reactions to spontaneous fission, and

the spontaneous fission rate *F*, the latter being the most important parameter. This leaves the detector efficiency undetermined and it needs to be predetermined experimentally, or by using alternative approaches such as assuming the sample multiplication to be known and then the detector efficiency can be unfolded.

Recently it was suggested that in addition to neutron multiplicity counting, gamma multiplicities be also used [4–6]. The motivation for using gamma counting is manifold: higher gamma multiplicity per fission, larger penetration through most of the strong neutron absorbers, and the relatively easy detection of gamma photons with organic scintillation detectors. The goal is still the same, i.e. to determine the above factors, plus the further unknowns introduced, such as the gamma leakage multiplication, the ratio γ of single gamma to fission gamma intensity and gamma detector efficiency. These can though be handled since three neutron and three gamma multiplicities can be measured simultaneously, so one has still as many unknowns as measured quantities.

However, there exists the further possibility of using the joint moments of the neutron and gamma counts, which supplies further independent measured data to determine still the same number of unknowns. Accounting also for the joint moments up to third order, there are altogether nine factorial moments. Hence the problem becomes overdetermined.

At the same time, the searched parameters are contained in a highly non-linear way in the multiplicity expressions. This is already true for the gamma moments and multiplicity rates alone. To handle the non-linearity of the problem which prevents an analytical inversion of the multiplicity rate formulae and in addition to make maximal use of the redundant information from the measurement when also the joint moments are used, the unfolding of the parameters has to be performed by least-square type unfolding methods. Actually, there is a conceptually simple non-parametric unfolding method for such a purpose, the artificial neural networks (ANNs), whose use will be demonstrated here.

In this paper we give the definitions of the quantities used and list all nine factorial moments. To give insight into the information contained in the joint moments, the dependence of the lowest order joint neutron-gamma moment on the non-leakage probability will be given quantitatively. In addition to the factorial moments, the multiplicity detection rates for the gamma photos will also be given. These have not been given previously. The derivation of converting the factorial moments into multiplicity detection rates follows the method described in a recent clarifying note on the relationship between factorial moments and the single, double and triple coincidence rates [10]. This will clearly show the relation between the measurable multiplicity rates and the factorial moments, when accounting for both multiple emission (spontaneous fission) and single emission (α,n) source events. In the last Section a description and test of the unfolding procedure is given by only using neutron multiplicity rates. Extension of the unfolding with ANNs to include gamma multiplicity rates will be reported in subsequent publications.

2. Definitions

The following definitions and conventions will be used. Random variables and their moments referring to neutrons will be denoted by ν, and those for gamma photons by μ. Variables referring to spontaneous fission will have a subscript sf, and those referring to induced fission a subscript *i*. For the factorial moments, there will always be a second index, giving the order of the moment. Hence, $\nu_{sf,2}$ will stand for $\langle\nu(\nu-1)\rangle$ in case of spontaneous fission. In addition, we will distinguish between two sets of variables for both neutrons and photons, depending on whether they belong to a source event, or to the total number of generated neutrons, which accounts for the internal multiplication (superfission [1]). The parameters belonging to the first set will be written with a subscript indication, such as $\nu_{s,2}$, ν_{i3}, whereas those of the second set will be denoted with just a numerical subscript indicating their order such as ν_1, μ_3.

The factorial moments corresponding to the distribution of neutrons or gammas emitted in fission, whether induced or spontaneous, are nuclear constants and are known in advance. However, as is usual in such work, it is practical to include the (unknown) contribution from generation of single

neutrons and photons, such as by (α, n) processes for the neutrons, into the moments related to spontaneous fission. Inclusion of single neutron generating processes in the calculations has long been applied. However there is a need for introducing a similar correction for gamma photons, since they also can be produced either in bunches (in the spontaneous fission process) or as singular gamma photons, in the same (α, n)-reactions which lead to the emission of single neutrons. In addition, there is also the presence of a "background" type emission of single gamma photons from neutron capture processes. The need for accounting such single photon generating processes was suggested by Sanchez [7]. In principle, there is also the possibility of producing individual gamma photons not only in the source processes, but also in the induced reactions, through inelastic scattering of the neutrons. Accounting for this possibility is though deferred to later work.

To account for the presence of single neutron producing events, one introduces the statistics of the total source events as a weighted average of the two processes [1],[6]. Quantities belonging to such a generalized source event will be denoted by a subscript s. Hence, we will use

$$\nu_{s,n} = \frac{\nu_{sf,n}(1+\alpha\delta_{1,n})}{1+\alpha\nu_{sf,1}} \tag{1}$$

as source moments for neutrons. Here Q_f and Q_α are the intensities of spontaneous fission and (α,n)-processes, respectively, and the factor α is defined as

$$\alpha = \frac{Q_\alpha}{Q_f \nu_{sf,1}}$$

For gamma photons produced also in connection to (α, n)-reactions, the source distribution of photons will change as:

$$f_s(n) = \frac{Q_\alpha}{Q_\alpha + Q_f}\delta_{n,1} + \frac{Q_f}{Q_\alpha + Q_f} f_{sf}(n) \tag{2}$$

This leads to the modified source moments defined as:

$$\mu_{s,n} = \frac{\mu_{sf,n} + \delta_{1,n}\alpha\nu_{sf,1}}{1+\alpha\nu_{sf,1}} \tag{3}$$

Here, $\nu_{sf,n}$ and $\mu_{sf,n}$ are the true moments of spontaneous fission (i.e. nuclear constants), whereas $\nu_{s,n}$ and $\mu_{s,n}$ are the ones corrected for the inclusion of production of neutrons and gammas by reactions other than fission. The moments relating to induced fission remain unchanged for neutrons and photons (ν_{in} and μ_{in} respectively).

The second set of moments and random variables concerns the distribution of the total number of generated or detected neutrons and gammas due to one initial *source* event, with internal multiplication included ("superfission" in Böhnels terminology [1]). These will not be denoted by any lettered subscript, only with a single number, expressing the order of the moment. The purpose of the calculations is to express these latter type of variables with the ones given by the nuclear constants, based on the distributions from spontaneous and induced fission, and the parameter *α*, describing the (unknown) relative intensity of production of single neutrons.

To obtain this relationship we need even the first collision probability p. For simplicity of the description, absorption will be neglected; however, detection efficiency will later be taken into account by the factors ε_n and ε_γ for the neutrons and photons, respectively. Absorption for the gamma photons can actually be included into the efficiency factor ε_γ in an exact way, due to the fact that gamma photons do not take part in the internal multiplication.

3. Factorial moments

Here we only list the various single and mixed factorial moments without any details of the underlying derivation. The principles of derivation through master equations can be found in [1], [4], [5] and [6].

3.1. Neutrons

First moments (singles)

$$\nu_1 = \mathbf{M}\,\nu_{s,1} \tag{4}$$

where

$$\mathbf{M} \equiv \frac{1-\mathrm{p}}{1-\mathrm{p}\,\nu_{i1}} \tag{5}$$

is called the *leakage multiplication*.

Second moments (doubles)

$$\nu_2 = \mathbf{M}^2 \left\{ \nu_{s,2} + \frac{\mathrm{p}}{1-\mathrm{p}\,\nu_{i1}} \nu_{s,1}\nu_{i2} \right\} = \mathbf{M}^2 \left\{ \nu_{s,2} + \frac{\mathbf{M}-1}{\nu_{i1}-1} \nu_{s,1}\,\nu_{i2} \right\}. \tag{6}$$

Third moments (triples)

$$\nu_3 = \mathbf{M}^3 \left\{ \nu_{s,3} + \frac{\mathbf{M}-1}{\nu_{i1}-1}(3\nu_{s,2}\nu_{i2} + \nu_{s,1}\,\nu_{i3}) + 3\left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right)^2 \nu_{s,1}\,\nu_{i2}^2 \right\}. \tag{7}$$

In these formulae the factorial moments of the combined source events are used. In order to be able to unfold the sample parameters, including the unknown factor α, one has to re-write these formulae in terms of the factorial moments of spontaneous fission and α. This is easily achieved by using Eq. (1), the results being

$$\nu_1 = \frac{\mathbf{M}}{(1+\alpha\nu_{sf,1})} \nu_{sf,1}(1+\alpha) \tag{8}$$

$$\nu_2 = \frac{\mathbf{M}^2}{(1+\alpha\nu_{sf,1})} \left[\nu_{sf,2} + \left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right) \nu_{sf,1}(1+\alpha)\nu_{i2} \right] \tag{9}$$

$$\nu_3 = \frac{\mathbf{M}^3}{(1+\alpha\nu_{sf,1})} \left[\nu_{sf,3} + \left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right) \left[3\nu_{sf,2}\nu_{i2} + \nu_{sf,1}(1+\alpha)\nu_{i3} \right] + 3\left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right)^2 \nu_{sf,1}(1+\alpha)\nu_{i2}^2 \right] \tag{10}$$

As discussed in [10], these formulae are incorrectly given in [8] and in several other publications. The multiplicity rates found in the literature including [8] are, however, correct.

3.2. Photons

The moments of the photons (which are more complicated due to the simple fact that photons do not self-multiply, rather they depend on the multiplication of neutrons), are given below. The detailed derivation can be found in [6].

Singles

$$\mu_1 = \mu_{s,1} + \frac{\nu_{s,1}\mathrm{p}\mu_{i1}}{1-\mathrm{p}\,\nu_{i1}} = \mu_{s,1} + \nu_{s,1}\mathbf{M}_\gamma \tag{11}$$

with

$$\mathbf{M}_\gamma \equiv \frac{\mathrm{p}\mu_{i1}}{1-\mathrm{p}\nu_{i1}} \tag{12}$$

being the *gamma (leakage) multiplication per one initial neutron*.

Doubles

$$\mu_2 = \mu_{s,2} + 2\mu_{s,1}\nu_{s,1}\mathbf{M}_\gamma + \nu_{s,2}\mathbf{M}_\gamma^2 + \nu_{s,1}\, g_2. \tag{13}$$

Triples

$$\mu_3 = \mu_{s,3} + 3\mu_{s,2}\nu_{s,1}\mathbf{M}_\gamma + 3\mu_{s,1}\{\nu_{s,2}\mathbf{M}_\gamma^2 + \nu_{s,1}\, g_2\} + \nu_{s,3}\mathbf{M}_\gamma^3 + 3\nu_{s,2}g_2 + \nu_{s,1}g_3. \tag{14}$$

The moments above are those of the source distribution, given by $\nu_{s,n}$ and $\mu_{s,n}$. Again, these have to be re-written in terms of the true fission neutron and gamma photon factorial moments and the factor *α* by equations (1) and (3). This leads to the expressions

$$\mu_1 = \frac{\mu_{sf,1} + \alpha\nu_{sf,1}}{(1+\alpha\nu_{sf,1})} + \frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma, \tag{15}$$

$$\mu_2 = \frac{\mu_{sf,2}}{1+\alpha\nu_{sf,1}} + 2\frac{(\mu_{sf,1}+\alpha\nu_{sf,1})}{(1+\alpha\nu_{sf,1})}\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma + \frac{\nu_{sf,2}}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma^2 + \frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}g_2 \tag{16}$$

and

$$\mu_3 = \frac{1}{(1+\alpha\nu_{sf,1})}\left[\mu_{sf,3} + 3\mu_{sf,2}\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma + \right.$$

$$\left. 3(\mu_{sf,1}+\alpha\nu_{sf,1})\left\{\frac{\nu_{sf,2}}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma^2 + \frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}g_2\right\} + \nu_{sf,3}\mathbf{M}_\gamma^3 + 3\nu_{sf,2}g_2 + \nu_{sf,1}(1+\alpha)g_3\right]. \tag{17}$$

One can note that the $(\alpha,\ n)$ processes affect also the photon equations, in a way analogous to the neutron moment equations, due to the single photon emission processes accompanying the $(\alpha,\ n)$ reactions. The appearance of the factor α becomes also highly non-linear, due to its occurring in a multiple way in the process. This also indicates the expected fact that an analytical inversion of the gamma multiplicity expressions for the sample parameters is not possible.

The factorial moments of the photon distribution initiated by a single neutron are defined in [6] as:

$$g_n = \left.\frac{d^3 g(z)}{dz^3}\right|_{z=1}. \tag{18}$$

This leads for g_2 and g_3 to the expressions

$$g_2 = \frac{\mathbf{M}-1}{\nu_{i1}-1}\{\mu_{i2} + 2\mu_{i1}\nu_{i1}\mathbf{M}_\gamma + \nu_{i2}\mathbf{M}_\gamma^2\} \tag{19}$$

and

$$g_3 = \frac{\mathbf{M}-1}{\nu_{i1}-1}\{\mu_{i3} + 3\mu_{i2}\nu_{i1}\mathbf{M}_\gamma + 3\mu_{i1}[\nu_{i2}\mathbf{M}_\gamma^2 + \nu_{i1}g_2] + \nu_{i3}\mathbf{M}_\gamma^3 + 3\nu_{i2}\mathbf{M}_\gamma g_2\}. \tag{20}$$

With no occurrences of $\nu_{s,n}$ and $\mu_{s,n}$, these expressions do not change for a compound source compared to a pure spontaneous fission source.

3.3. Mixed moments

Instead of the first mixed moment, we give the covariance:

$$\mathbf{Cov}\{\nu,\mu\} \equiv \mathrm{E}\{\nu\mu\} - \mathrm{E}\{\nu\}\,\mathrm{E}\{\mu\} = \left\{(\nu_{s,2}-\nu_{s,1}^2)\mathbf{M}\mathbf{M}_\gamma + \nu_{s,1}\frac{\mathbf{M}-1}{\nu_{i1}-1}\{\mu_{i1}\nu_{i1}\mathbf{M} + \nu_{i2}\mathbf{M}\mathbf{M}_\gamma\}\right\}. \tag{21}$$

For illustration, the dependence of the coherence on the first collision probability p is shown in Fig. 1.

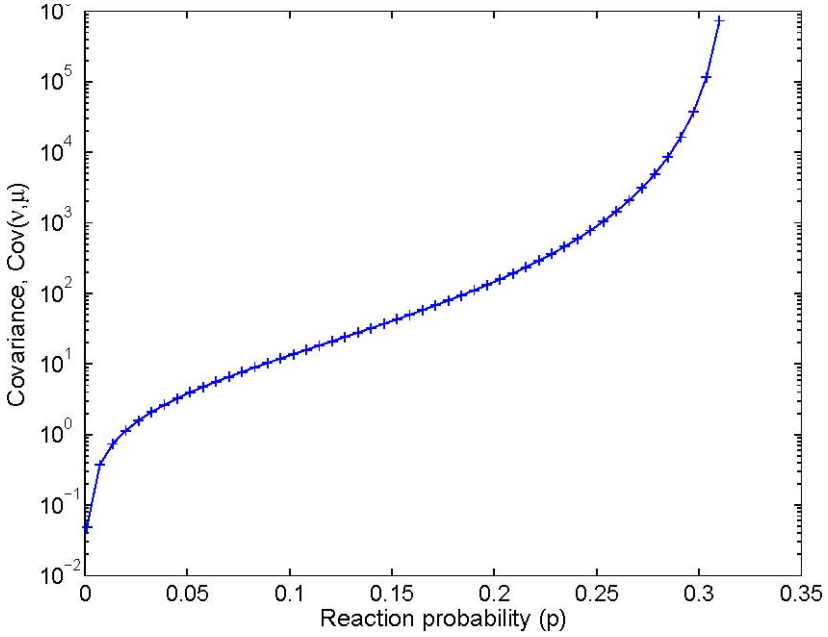

Figure 1: Covariance between neutrons and photons.

Higher moments:

$$\begin{aligned}\left\langle \nu(\nu-1)\,\mu\right\rangle &= \left\{\mu_{s,1}\left[\nu_{s,2}\mathbf{M}^2+\nu_{s,1}h_2\right]+\nu_{s,3}\mathbf{M}_\gamma\,\mathbf{M}^2+\right.\\ \left\langle n(n-1)\,m\right\rangle &= \left\{m_{s,1}\left[n_{s,2}\mathbf{M}^2+n_{s,1}h_2\right]+n_{s,3}\mathbf{M}_g\,\mathbf{M}^2+\right.\\ &\quad \left.\left. n_{s,2}\left[2\mathbf{M}c_{1,1}+\mathbf{M}_g\,h_2\right]+n_{s,1}c_{2,1}\right\}\right.\end{aligned} \tag{22}$$

and

$$\begin{aligned}\left\langle \nu\ \mu(\mu-1)\right\rangle &= \left\{\mu_{s,2}\nu_{s,2}\mathbf{M}+2\mu_{s,1}\left[\nu_{s,2}\mathbf{M}\,\mathbf{M}_\gamma+\nu_{s,1}c_{1,1}\right]+\right.\\ &\quad \left.+\nu_{s,3}\mathbf{M}\mathbf{M}_\gamma^2+\nu_{s,2}\left[g_2\mathbf{M}+2\mathbf{M}_\gamma\,c_{1,1}\right]+\nu_{s,1}c_{1,2}\right\}\end{aligned} \tag{23}$$

In the above,

$$c_{1,1}=\left\{\mathrm{p}\frac{\mu_{i1}\nu_{i1}\mathbf{M}+\nu_{i2}\mathbf{M}\mathbf{M}_\gamma}{1-p\nu_{i1}}\right\}, \tag{24}$$

$$c_{2,1}\ \equiv\frac{\mathrm{p}}{1-\mathrm{p}\,\nu_{i1}}\{\mu_{r,1}[\nu_{i2}\,\mathbf{M}^2+\nu_{i1}\,h_2]+\nu_{i3}\mathbf{M}^2\mathbf{M}_\gamma+\nu_{i2}[h_2\mathbf{M}_\gamma+2\mathbf{M}c_{1,1}]\}, \tag{25}$$

$$c_{1,2}\ \equiv\frac{\mathrm{p}}{1-\mathrm{p}\,\nu_{i1}}\{\mu_{i2}\nu_{i1}\mathbf{M}+2\mu_{i1}[\nu_{i2}\mathbf{M}\mathbf{M}_\gamma+\nu_{i1}c_{1,1}]+\nu_{i3}\mathbf{M}\mathbf{M}_\gamma^2\ +\nu_{i2}[2\mathbf{M}_\gamma c_{1,1}+\mathbf{M}g_2]\}. \tag{26}$$

Here h_2 is the second factorial moment of the total number of neutrons generated in the sample, internal multiplication included, by *one single neutron*. This and other moments can be found in [4]:

$$h_2=\frac{\mathbf{M}-1}{\nu_{i1}-1}\nu_{i2}\mathbf{M}^2. \tag{27}$$

While these formulae are correct for a pure spontaneous fission source, they need to be expanded using Eqs. (1) and (3), to account properly also for the effect of the $(\alpha,\ n)$ process on the single neutron and gamma photon generating processes. The following expressions are then obtained:

$$\mathbf{Cov}\{\nu,\mu\}=\frac{1}{(1+\alpha\nu_{sf,1})}\left\{(\mu_{sf,1}+\alpha\nu_{sf,1})\frac{(\nu_{sf,1}(1+\alpha))}{(1+\alpha\nu_{sf,1})}\mathbf{M}+\right.$$

$$+\nu_{sf,2}\mathbf{M}\mathbf{M}_{\gamma}+\nu_{sf,1}(1+\alpha)\frac{\mathbf{M}-1}{\nu_{i1}-1}\{\mu_{i1}\nu_{i1}\mathbf{M}+\nu_{i2}\mathbf{M}\mathbf{M}_{\gamma}\}\Bigg\} \quad (28)$$

$$\langle\nu(\nu-1)\rangle\mu=\frac{1}{(1+\alpha\nu_{sf,1})}\Bigg\{(\mu_{sf,1}+\alpha\nu_{sf,1})\left[\frac{\nu_{sf,2}}{(1+\alpha\nu_{sf,1})}\mathbf{M}^2+\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}h_2\right]+$$

$$\nu_{sf,3}\mathbf{M}_{\gamma}\ \mathbf{M}^2+\nu_{sf,2}\left[2\mathbf{M}c_{1,1}+\mathbf{M}_{\gamma}\,h_2\right]+\nu_{sf,1}(1+\alpha)c_{2,1}\Bigg\} \quad (29)$$

and

$$\langle\nu\ \mu(\mu-1)\rangle=$$

$$=\frac{1}{(1+\alpha\nu_{sf,1})}\Bigg\{\mu_{sf,2}\frac{\nu_{sf,2}}{(1+\alpha\nu_{sf,1})}\mathbf{M}+2(\mu_{s,1}+\alpha\nu_{sf,1})\left[\frac{\nu_{sf,2}}{(1+\alpha\nu_{sf,1})}\mathbf{M}\,\mathbf{M}_{\gamma}+\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}c_{1,1}\right]+$$

$$+\nu_{sf,3}\mathbf{M}\mathbf{M}_{\gamma}^2+\nu_{\ sf,2}\left[g_2\mathbf{M}+2\mathbf{M}_{\gamma}\,c_{1,1}\right]+\nu_{sf,1}(1+\alpha)c_{1,2}\Bigg\} \quad (30)$$

4. Multiplicity detection rates

The measured quantities are the multiplicity rates. To convert the factorial moments of a single source event into detection rates of multiplicities, one has to account for the intensity of the source events and the detection efficiency. The effect of the finite measurement gate time in multiple coincidence measurements, quantified with the relative gate width factors as described in [8], will be omitted here.

Measurable quantities - Neutrons

To find the measurable quantities such as singles, doubles etc., one needs first to find the factorial moments $\tilde{\nu}_k$ of the *detected* neutrons per one initial event. The reason this quantity is needed is that e.g. a measured doublet could be the result of detecting two particles from a higher order multiplet. This further requires the introduction of the detector efficiency and for the first few moments these factorial moments are defined as:

$$\tilde{\nu}_1=\varepsilon\nu_1,$$
$$\tilde{\nu}_2=\varepsilon^2\nu_2,$$
$$\tilde{\nu}_3=\varepsilon^3\nu_3.$$

Now for neutrons the detection *rates* are also related to the total source intensity. Using C_k as the notation for the *k*-th order multiplet (such that $C_1=S$, $C_2=\mathrm{D}$ etc) and the total neutron source rate $Q_n\equiv Q_f+Q_\alpha$ it follows that

$$C_k=Q_n\left\langle\binom{n}{k}\right\rangle=Q_n\sum_n\frac{n!}{k!(n-k)!}P(n)=Q_n\frac{\tilde{\nu}_k}{k!}. \quad (31)$$

Using the *α*-factor the source factor can be expressed as:

$$Q_n=F+Q_\alpha=F(1+\alpha\nu_{sf,1}). \quad (32)$$

In the case of singles for neutrons the following expression is derived:

$$S=F\varepsilon_n(1+\alpha\nu_{sf,1})\frac{\mathbf{M}\,\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}=F\varepsilon_n\mathbf{M}\,\nu_{sf,1}(1+\alpha). \quad (33)$$

Note how the scaling factor between the fission source and the total source intensity cancels out in the expression for the measurable singles. In a similar way doubles and triples can be derived as:

$$D=\varepsilon_n^2C_2=\frac{F\varepsilon_n^2\mathbf{M}^2}{2}\left[\nu_{sf,2}+\left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right)\nu_{sf,1}(1+\alpha)\nu_{i2}\right], \quad (34)$$

$$T=\varepsilon_n^3 C_3=\frac{F\varepsilon_n^3\mathbf{M}^3}{6}\left\{\nu_{sf,3}+\left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right)[3\nu_{sf,2}\nu_{i2}+\nu_{sf,1}(1+\alpha)\nu_{i3}]\right.$$

$$\left.+3\left(\frac{\mathbf{M}-1}{\nu_{i1}-1}\right)^2\nu_{sf,1}(1+\alpha)\nu_{i2}^2\right\}. \qquad (35)$$

These are the quantities one measures in multiplicity counters. It is these expressions that serve as the basis for the different approaches to find the various unknown parameters, as described in [3], Most commonly one assumes the neutron detector efficiency ε_n to be known, and solve the equations for fission rate (mass), F , leakage multiplication, M , and α - the relative contribution from single-neutron sources.

Measurable quantities - Photons

In the case of photons, the moments are considerably more complicated due to accounting for both neutrons and photons. It is still possible to derive equations for the measurable quantities of singles, doubles and triples, in a manner similar to that of neutrons.

The modified moments account also for single emitted neutrons and photons, equations (1) – (3). These will lead to lengthier expressions. In addition, when accounting for the effect of all source events, for photons one has to account for the possibility of a single photon source which is not connected to the neutron chain. This can be made in a way analogous to the accounting for the (*α,* n) processes for neutrons. Defining *γ* as the ratio between the single photon source strength, Q_γ , and the neutrons source strength, Q_n , the gamma singles doubles and triples can be expressed as

$$S=\left[\gamma F(1+\alpha\nu_{sf,1})+F(1+\alpha\nu_{sf,1})\tilde{\mu}_1\right],$$

$$D=F(1+\alpha\nu_{sf,1})\frac{\tilde{\mu}_2}{2},$$

$$T=F(1+\alpha\nu_{sf,1})\frac{\tilde{\mu}_3}{3!}.$$

Also here we used the tilde notation for the detected moments, i.e. accounting for a detector efficiency, ε_γ in the case of gamma photons. With the previous formulae one can list the full expressions for the singles, doubles and triples of photons. In the simplest case of singles one has:

$$S=\varepsilon_\gamma\left[\gamma F(1+\alpha\nu_{sf,1})+F(1+\alpha\nu_{sf,1})\left\{\frac{\mu_{sf,1}+\alpha\nu_{sf,1}}{(1+\alpha\nu_{sf,1})}+\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma\right\}\right]=$$

$$=F\varepsilon_\gamma\left[\gamma(1+\alpha\nu_{sf,1})+\mathbf{M}_\gamma\left\{\mu_{sf,1}+\alpha\nu_{sf,1}+\nu_{sf,1}(1+\alpha)\right\}\right]. \qquad (36)$$

For doubles and triples the expressions grow longer:

$$D=\frac{\varepsilon_\gamma^2 F}{2}\left[\mu_{sf,2}+2(\mu_{sf,1}+\alpha\nu_{sf,1})\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma+\nu_{sf,2}\mathbf{M}_\gamma^2+\nu_{sf,1}(1+\alpha)\,g_2\right] \qquad (37)$$

$$T=\frac{\varepsilon_\gamma^3 F}{6}\left[\mu_{sf,3}+3\mu_{sf,2}\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma+3(\mu_{sf,1}+\alpha\nu_{sf,1})\left\{\frac{\nu_{sf,2}}{(1+\alpha\nu_{sf,1})}\mathbf{M}_\gamma^2+\frac{\nu_{sf,1}(1+\alpha)}{(1+\alpha\nu_{sf,1})}g_2\right\}\right.$$

$$\left.+\nu_{sf,3}\mathbf{M}_\gamma^3+3\nu_{sf,2}g_2+\nu_{sf,1}(1+\alpha)g_3\right]. \qquad (38)$$

Our proposal is to use these equations much in the same way as those for neutrons are used to find sample parameters. What is new here is the fraction γ of single photons in the source events, and the presence of the detector efficiency for photons, which could be pre-calibrated. The gamma leakage

multiplication M_γ on the on other the hand is unknown much like the case of the leakage multiplication of neutrons (M).

The unknown parameters M, M_γ, α, γ, ε_n and ε_γ can be determined from the above equations with a nonlinear least squares non-parametric fitting to the measured values. In the present model M and M_γ are not two independent unknowns since both contain only one unknown parameter, the first collision probability p, hence the number of unknowns is even fewer. The possibilities of the unfolding of the unknowns will be discussed below.

5. Application

As mentioned previously, unlike for the neutron expressions, the complexity of the expressions for the gamma photons prevents the possibility of using analytical inversion of the multiplicity rate expressions. Hence we propose the use of artificial neural network (ANN) techniques for the unfolding of sample parameters from the measured multiplicity rates.

The use of ANNs can be tested already on the known case of neutron multiplicities, which can serve also as a first test. In addition, it offers some advantages already for this relatively simple case. Namely, the analytical inversion is only possible as long as only three unknowns are attempted to be retrieved from the three multiplicity rates. This has the effect that the neutron efficiency needs to be known in advance. With ANN techniques, there is a larger flexibility, since ANNs can utilize the rich information in the non-linearity of the expressions to unfold more parameters than the number of expressions. Hence there is a chance that in addition to the usual three parameters, also the detector efficiency can be retrieved.

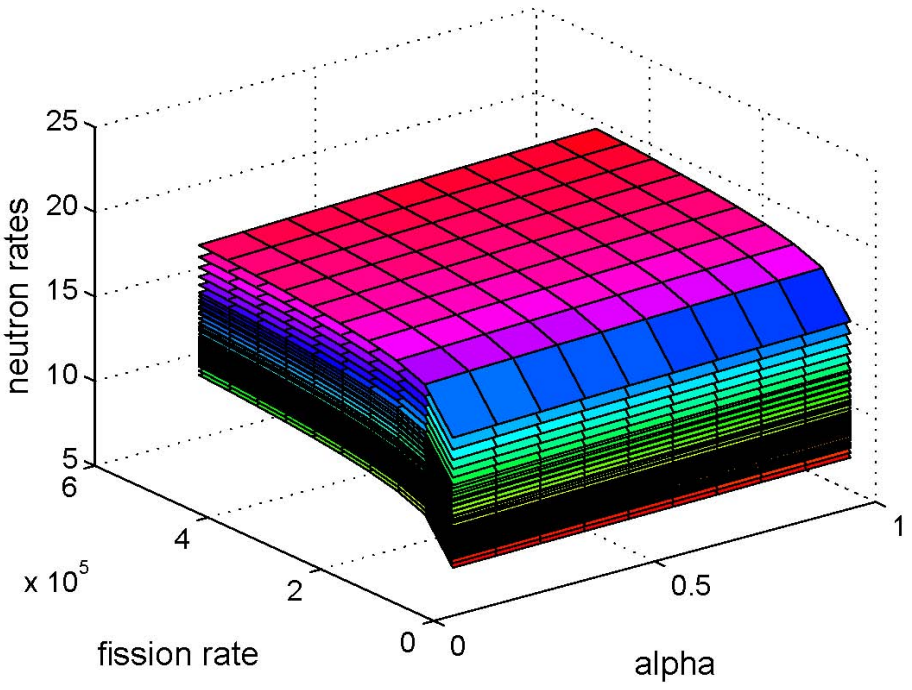

Figure 2. An example of the training data used which are calculated for different values of F, p and α.

Some initial tests were made to unfold the three usual sample parameters from the neutron singles, doubles and triples rates by ANN methods. The analytical expressions were used, by sweeping with the parameters F, M and α over realistically possible values, to generate input patterns for the training of a simple feedforward backward propagation network with three inputs and three outputs. Two hidden layers were needed for successful training, but the structure of the network may be refined more in future work.

Below some results are shown, obtained after the initial training of the network was completed. The trained network was tested by further sample vectors generated the same way as the training set. The dependence of the training data on the input parameters is shown, for the case of doubles, in Fig. 2.

	fission rate (F)	α	p
max. rel. error (%)	0.0000	0.0021	0.0001
min. rel. error (%)	-0.0001	-0.0019	-0.0001

Table 1. Preliminary training results of ANN, using the neutron equations to simulating large plutonium samples in the kg-range.

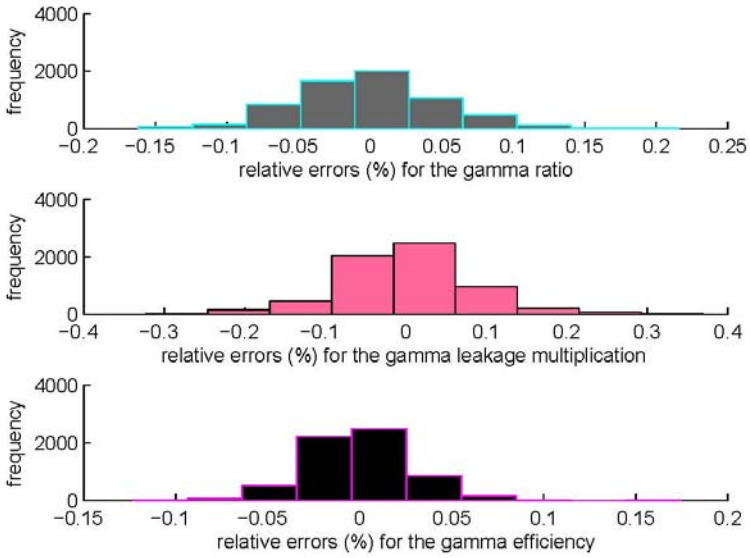

Figure 3. Histogram showing the distribution of the errors of the predicted parameters: *F*, p and *α*.

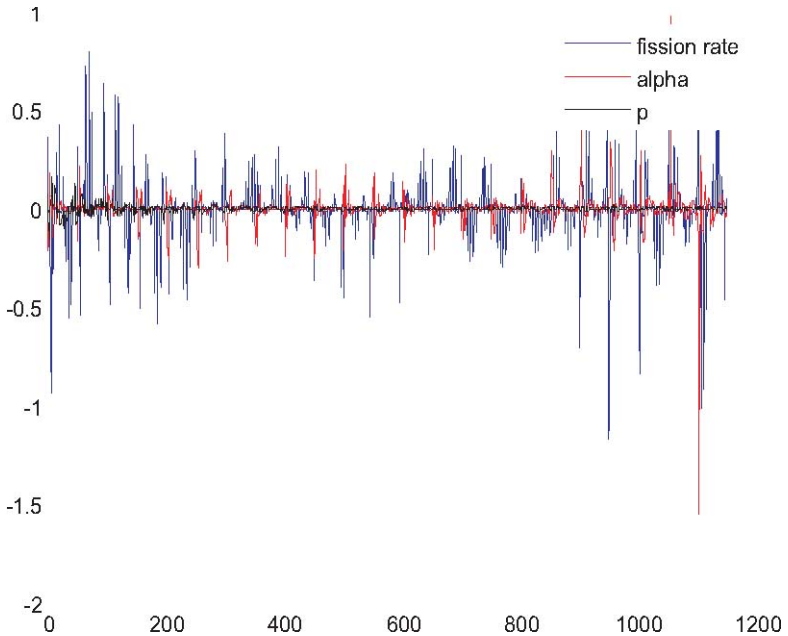

Figure 4. Relative errors after preliminary training shown for the parameters investigated: F , p and α .

Preliminary results show that the parameters F , α , p can be evaluated with the relative errors less smaller than 0.001%. These results are very promising. Figure 3 and 4 shows the variation of the error associated with the variables. The work is progressing fast with testing the possibility of determining more parameters with various combinations of neutron, photon and joint multiplicities.

Table 2 shows the relative error when using the three mixed moments only as input. to try and unfold four parameters. As can be seen, the accuracy is good also for underdetermined systems.

To make a better comparison with how real measurement data which have larger uncertainties would affect the ANN's performance we did training on the neural network, with and added 10% of "noise" to the input data. To simulate the scatter in measured data. What can be seen in Fig 5, is that the uncertainties for triples induces the largest error in the predicted data. However, also when training with noise data, the ANN performs very satisfactorily, which is a good indicator of its applicability.

	α	M	ε_n	ε_γ
max	1.7353	0.1294	0.0288	0.0111
min	-0.2978	-0.5432	-0.0132	-0.0140

Table 2. Accuracy of the ANN when using the 3 mixed moments to unfold 4 parameters.

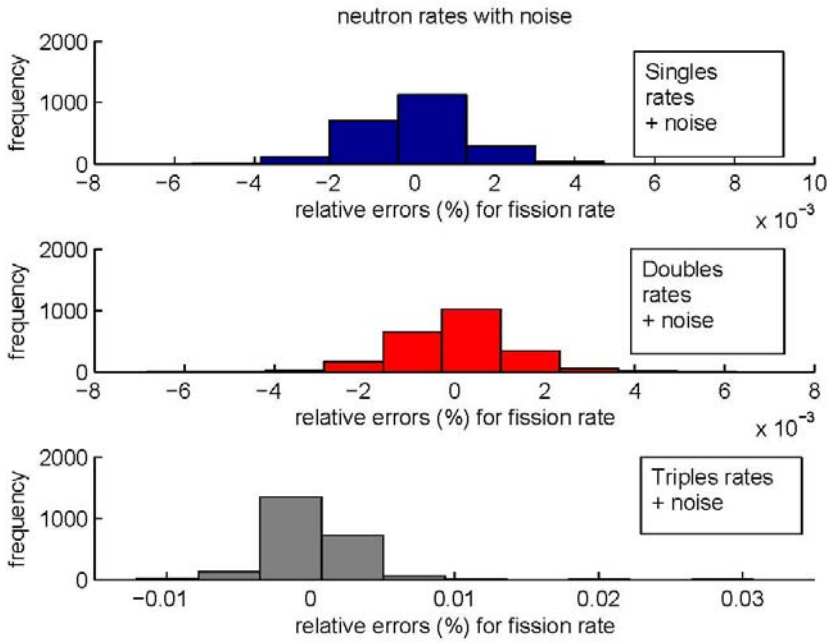

Figure 5. Investigation of the errors in multiplicity rates induced by 10% noise added to the training data.

6 Conclusions

The present paper shows that by taking all possible auto- and cross factorial moments of the neutron and gamma counts into account, one has nine expressions which are functions of five independent parameters. The generation of single photons by various processes, in addition to the multiple ones from fissions, was considered after a suggestion of R. Sanchez.

It is suggested that these multiplicity rates be inverted by non-linear non-parametric least squares methods, namely with the use of artificial neural networks, to which the above equations can be used to generate training data. Final validation and further development of the ANN is ongoing. The results are very promising and of good accuracy. When adding noise to the training data to simulate

measurement uncertainties, the induced uncertainties for the ANN can be kept very low. The training and performance of ANN's using all moments for both neutrons and photons is computationally more demanding, but still within manageable range. Another advantage could be that when using Monte-Carlo simulations to generate training data, the network might also adapt to the statistical uncertainties in the input data, which are reconstructed by the simulations in a realistic manner. Such effects cannot be accounted for by analytical inversion methods.

7. Acknowledgements

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Simulated Verification of Fuel Element Inventory in a Small Reactor Core Using the Nuclear Materials Identification System (NMIS)

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Abstract:

The International Panel on Climate Change projects that by 2050 the world energy demand may double. Although the primary focus for new nuclear power plants in industrialized nations is on large plants in the 1000–1600 MWe range, there is an increasing demand for small and medium reactors (SMRs). About half of the innovative SMR concepts are small (<300 MWe) reactors with a 5–30 year life without onsite refueling, also known as battery-type reactors. These reactors are particularly attractive in countries with small power grids and for nonelectrical purposes such as heating, hydrogen production, and seawater desalination. Traditionally, this size of reactor has been used for nautical propulsion. It is designed as a permanently sealed unit to prevent material diversion of the uranium in the core by the user. However, after initial fabrication, it will be necessary to verify that the newly fabricated reactor core contains the declared quantity of uranium to thwart material diversion by the builder. The Nuclear Materials Identification System (NMIS) with fast neutron imaging uses active interrogation and a fast time correlation processor to characterize fissile material. This paper describes preliminary evaluations of the feasibility of using the NMIS to validate the amount of fissile material and design of the completed core. The MCNP-PoliMi computer code was used to simulate NMIS measurements of a small, sealed reactor core. Because most battery-type reactor designs are still in the early design phase, the simulations used a Russian icebreaker core that is already in production. These simulations show how the radiographic capabilities of NMIS could be used to detect the diversion of fissile material by detecting void areas in the assembled core where fuel elements have been removed. The simulations have shown that NMIS fast neutron imaging can detect the removal of as little as 1% of the fuel inventory from a single location or the removal of 3.7% (and probably much less) throughout the reactor core.

Keywords: NMIS, MCNP-PoliMi, neutron, radiography, SMR

1. Introduction

The world demand for energy is continually increasing. By 2050, the International Panel on Climate Change projects that it may double [1]. As energy demand increases, nuclear energy is expected to provide a significant portion of new energy production. In developed nations, the economy of scale dictates that most of these new reactors will be in the 1000+ megawatt electric (MWe) range. In many smaller, developing nations, however, reactors of this size are not practical because the initial cost is too great or the country lacks the power distribution infrastructure necessary to accommodate such a large reactor. A large reactor also may be inappropriate for uses other than energy production, such as seawater desalinization or heating. A small or medium reactor (SMR) in the <700 MWe range may be more appropriate for these types of applications [2].

A 2006 International Atomic Energy Agency report [3] indicates that about half of the innovative new SMR designs are small (<300 MWe) reactors that are not refueled onsite. These reactor types feature a longer (5 to 30 year) interval between refuelings and are referred to as battery-type reactors because refueling would be accomplished by replacing the entire core [2]. Although this type of core is delivered

from the assembly plant as a sealed unit, it will be necessary for inspectors to verify that material has not been diverted during the assembly of the core. The Nuclear Materials Identification System (NMIS) with imaging can be used to verify the uranium fuel inventory and enrichment nonintrusively. Verification could be accomplished by combining measurement of the physical parameters of the core (height, weight, radius, etc.) with NMIS measurements to estimate the enrichment of the core using multiplicities and a density map using fast neutron tomography. A simultaneous visual inspection of the core would verify the containment to ensure no material was diverted at a later time.

This initial scan of the first core could then be used to build a measurement template for nonsensitive cores. The inventory of future cores could then be verified more quickly using a template matching method [4] and an inspection of containment. For sensitive core designs, these measurements could be conducted with some acceptable method to hide sensitive design information from inspectors while still verifying the correct quantity of fissile material is in the core.

This series of preliminary Monte Carlo simulations will attempt to quantify the accuracy with which the fast neutron radiography capability of NMIS can verify the fuel inventory in a small reactor core. Only the neutron radiography aspect will be considered in this paper, but it is assumed that a simultaneous measurement of source-correlated multiplets from which the multiplicity and Feynman variance can be obtained is simultaneously verifying the enrichment of the fuel. Various scenarios in which a quantity of fuel has been removed will be compared with a baseline case to see if NMIS can conclusively determine that fuel has been diverted. If the resulting attenuation plot differs from the baseline with a confidence of > 99% NMIS will be considered to have successfully thwarted the material diversion.

2. The Nuclear Materials Identification System with Imaging

NMIS was developed at the Oak Ridge National Laboratory (ORNL) and the Oak Ridge Y-12 Plant for identifying and characterizing fissile materials [5]. The NMIS processor computes fast time correlations between two or more detectors in real time. In recent years, both radiographic and tomographic imaging capabilities have been added to NMIS [6,7].

The primary component of NMIS is the fast time-correlation processor. Most of the processing is performed by a commercial off-the-shelf rack-mounted computer system. Two PCI add-on boards allow the processor to sample 10 channels of data at a rate of up to 1 gigahertz per channel. Eight or more detector signals can be combined on a single channel by assigning each detector signal a unique pulse width that can be identified by the NMIS software. A special software suite written at ORNL processes incoming detector data and computes the detector-detector correlations in real time.

When used in active interrogation mode with a source that can be time-tagged, the NMIS processor can measure the time of flight between the source and other detectors. Using a known source-to-detector distance, this time-of-flight measurement can be used to separate directly transmitted radiation generated by the source from background radiation or scattered particles. By using several detectors arrayed in an arc equidistant from the radiation source, NMIS can determine the approximate path that each interrogating particle took to the detector. This information can then be used to reconstruct a radiograph of an object between the source and detectors. By rotating the target and making projections at several different angles, a full 3-dimensional tomographic reconstruction can be computed.

The source most commonly used for NMIS imaging measurements is an associated particle deuterium-tritium (DT) neutron generator. This generator produces monoenergetic 14.1 MeV neutrons via the $^{3}H(^{2}H,n)^{4}He$ reaction. Because the DT neutrons are monoenergetic, they produce an easily identifiable peak in the time-of-flight spectrum. Using fast plastic scintillators made of EJ-230 (BC-420) material, the full width at half maximum of this peak is less than 5 ns. Most of this 5 ns width comes from the time of flight of the 14.1 MeV neutrons through the 15.24 cm depth of the detectors, which produces a 3 ns spread. The DT neutrons are time-tagged by an alpha particle detector integrated into the neutron generator. Because the alpha particle and neutron are produced back-to-back in the center-of-mass coordinate system, the alpha particle detector also provides electronic collimation of the neutrons into a cone approximately 180° opposite the direction of the alpha particles [8].

The geometry of the DT generator and the alpha particle detector limits the tagged neutrons to a cone with an opening angle of 45°. The cone can be further limited by placing an opaque mask between the alpha detector and the photomultiplier tube. The most commonly employed mask limits the neutron cone to a fan 10° high by 45° wide. A fan of this size occupies approximately 1/288th of the total solid angle. With an isotropic source strength of 3×10^7 neutrons per second and an alpha detector efficiency of 0.85, approximately 90,000 correlated neutrons per second are produced within the fan boundaries. The alpha particle detector can also be coupled to a pixelated photomultiplier tube [9]. This configuration allows for better angular collimation of the DT neutron beam without reducing the number of tagged neutrons. The pixelization of the neutron fan beam reduces scattering effects to produce improved neutron radiographs with better spatial resolution.

3. MCNP-PoliMi

The MCNP-PoliMi code was used to perform the simulations in this paper. MCNP-PoliMi is a modification of the standard MCNP 4c code developed at the Polytechnic of Milan (PoliMi) by Sara Pozzi and Enrico Padovani [10]. Unlike standard MCNP codes which model average particle behaviors, MCNP-PoliMi attempts to model each neutron–nucleus interaction as accurately as possible. In looking at correlations between pairs of detectors, this is necessary because simply using average values for each interaction can produce incorrect results [11].

Another useful feature of MCNP-PoliMi is that it produces a specially modified PTRAC file containing each collision in the detector cells along with the type of collision, the energy imparted, and other useful data. A post-processor can then convert this file into a detector response explicitly by calculating the light output for each interaction using light curves based on experimental results. The post-processor also applies detector deadtimes and other attributes of the physical detectors. In this manner, the light pulses in the detector are modeled as accurately as possible. The post-processor then uses these pulses to compute source-detector and detector-detector correlations.

4. Reactor Core Model

The reactor core used in these simulations is based on a design for a Russian icebreaker core from A. C. Diakov et al. [12]. This reactor was selected because small reactors were first used as power plants for naval vessels [2], and thus battery-type reactors could be expected to at least loosely resemble such a design. To simplify the model for MCNP, the design was modified somewhat. In particular, the fuel pins were converted from a cross shape to a standard cylindrical fuel pin shape with an equivalent volume to simplify the model and reduce simulation times. The initial MCNP input deck of the reactor core was developed by F. Dalnoki-Veress, A. Glaser, and F. Von Hippel of the Program on Science and Global Security at Princeton University [13]. The core uses an aluminum-uranium metal alloy fuel with a ratio of approximately 9.6 aluminum atoms per uranium atom and a uranium enrichment of 40%. Table 1 shows the parameters of the modeled reactor core, and Figure 1 shows a cross-section view of the simplified core.

Active Core Height	90 cm
Core Diameter	90 cm
Fuel Pin Diameter	0.4026 cm
Cladding Thickness	0.0586 cm
Fuel Pin Pitch	0.7 cm
Fuel Volume Fraction	0.3
Cladding Volume Fraction	0.2
Fuel Density	4.573 g/cm^3
Fuel Composition	U-Al$_{9.6}$
Uranium Enrichment	40%
Total Uranium Mass	375 kg
Total ^{235}U Mass	150 kg

Table 1. Parameters of the modeled reactor core in its baseline configuration.

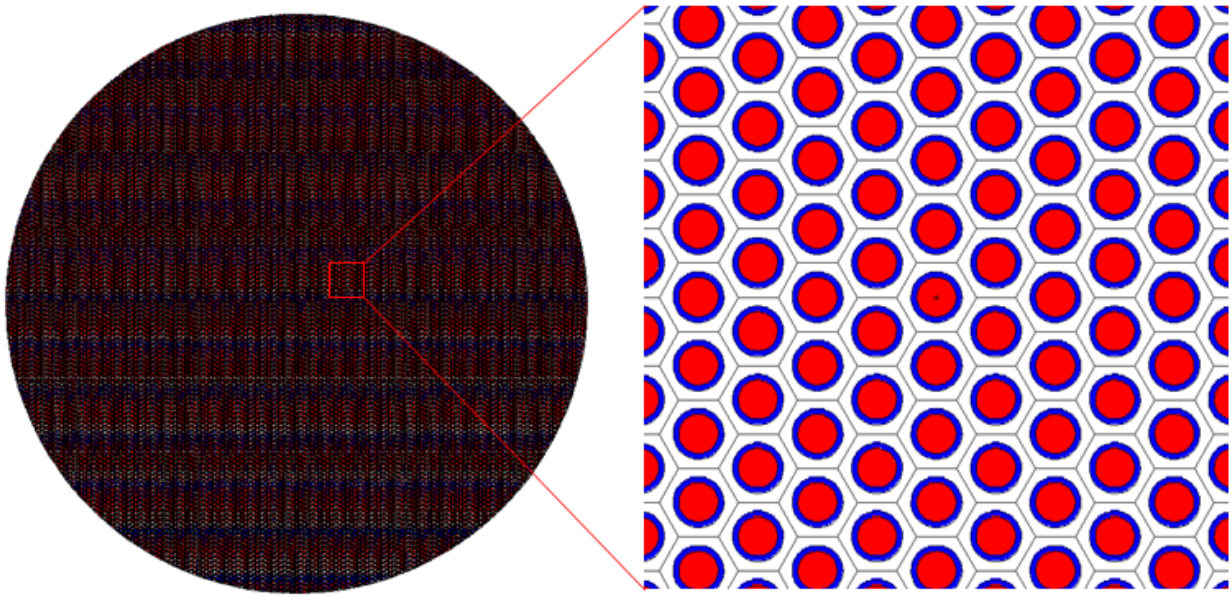

Figure 1. A cross-sectional view of the baseline reactor core model (left) and a close-up view of a 6 by 6 cm region of the core. Note that the hexagonal lattice structure is for reference only and does not represent physical surfaces in the core.

The DT neutrons were modeled as a monoenergetic 14.1 MeV neutron point source. To simulate the electronic collimation of the alpha detector, the neutrons were limited to a fan 10° high and 45° wide. An array of 32 plastic 2.54 × 2.54 × 15.24 cm was modeled along a 218.44 cm radius arc equidistant from the DT neutron source. This source-to-detector distance matches the radius of one of the existing NMIS imaging arms. The detectors were spaced with their centers 5.60 cm apart in order to completely cover the 45° horizontal extent of the arc. The source and detectors were placed in the model so that the reactor core was located between the source and detector array and centered both horizontally and vertically on the DT neutron fan. The center of the reactor core was placed 135 cm from the source so that the entire reactor fit within the DT neutron fan. Figure 2 shows the layout of the DT generator, reactor core, and detector array with the neutron fan shown.

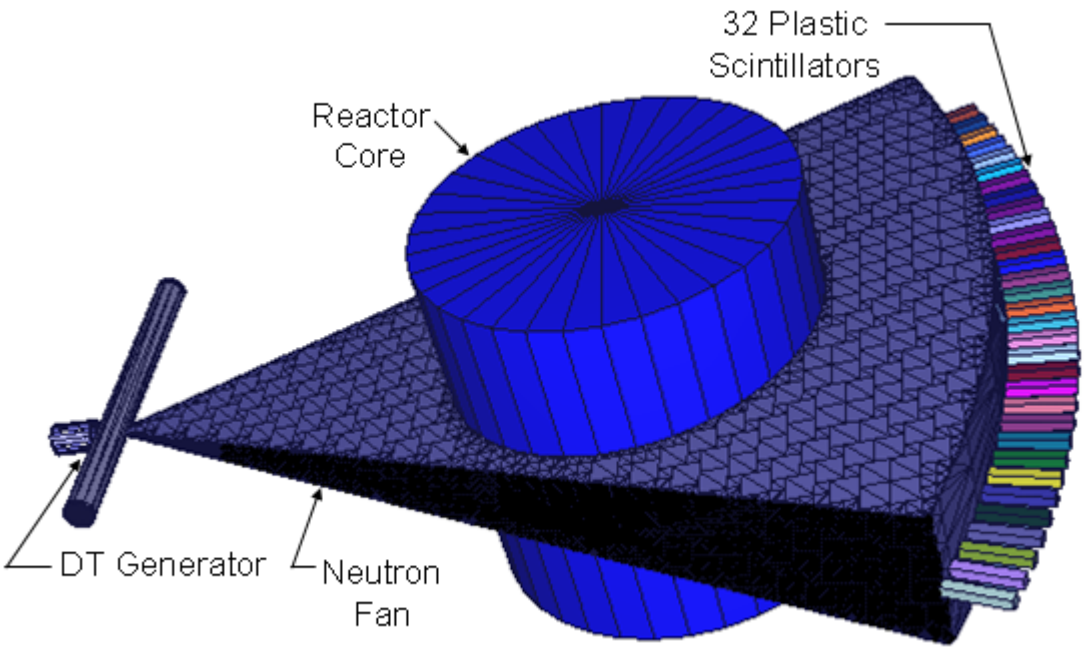

Figure 2. The configuration of the NMIS radiography measurements. A 45° wide neutron fan passes through the reactor core to fast plastic scintillators which will measure the transmission as a function of angle. The neutron fan shows the paths of source neutrons and does not represent a physical structure.

Each simulated measurement consisted of four simulations. In each simulation, the detector arm was rotated slightly (±0.125, ±0.375 times the detector-detector separation distance) to increase the angular resolution of the detector array. This process is referred to as sub-sampling the detectors. Each simulation consisted of 5.3×10^7 source neutrons. This corresponds to a measurement time of approximately 10 minutes for a DT generator with a source strength of 3×10^7 neutrons per second emitted isotropically. With four sub-samples, each simulated measurement corresponds to a 40 minute measurement with the current NMIS imaging system. Future DT generators with an increased neutron output (10^8 n/s or more) are expected to reduce this measurement time by a factor of at lease 3.

The transmission of 14.1 MeV neutrons was calculated by integrating the total number of correlations in a ±2 ns window around the 14.1 MeV timing peak. The attenuation at each detector position was then calculated by the equation

$$\frac{I}{I_0} = e^{-\tau} \qquad \text{(Equation 1)}$$

where τ is the attenuation in mean free paths, I is the number of correlations per source neutron in the simulation, and I_0 is the number of correlations per source neutron in a simulation with no reactor. Solving Equation 1 for τ yields

$$\tau = -\ln\left(\frac{I}{I_0}\right) \qquad \text{(Equation 2)}$$

5. Simulations

5.1. Scenario 1

To estimate the minimum detectible amount of removed fuel, several fuel pins were removed from the reactor core. The attenuation as a function of detector angle was then compared with results from a

reactor with no voids (the baseline). In the first group of simulations, this was accomplished by defining a cylindrical void at the center of the reactor to represent a bulk removal of fuel pins. The radius of the void was varied to quantify the sensitivity of the NMIS imaging. Table 2 shows the parameters of the modeled cases. Figure 3 shows a model of a core with a 9 cm radius void in the center.

Title	Void Radius (cm)	Void Fraction (%)	Missing Mass ^{235}U (kg)
Baseline	None	0	0
DTo1	3	0.40	0.67
DTo2	6	1.78	2.67
DTo3	9	4.00	6.00
DTo6	18	16.0	24.0
DTo9	27	36.0	54.0

Table 2. Parameters used for simulations in scenario 1.

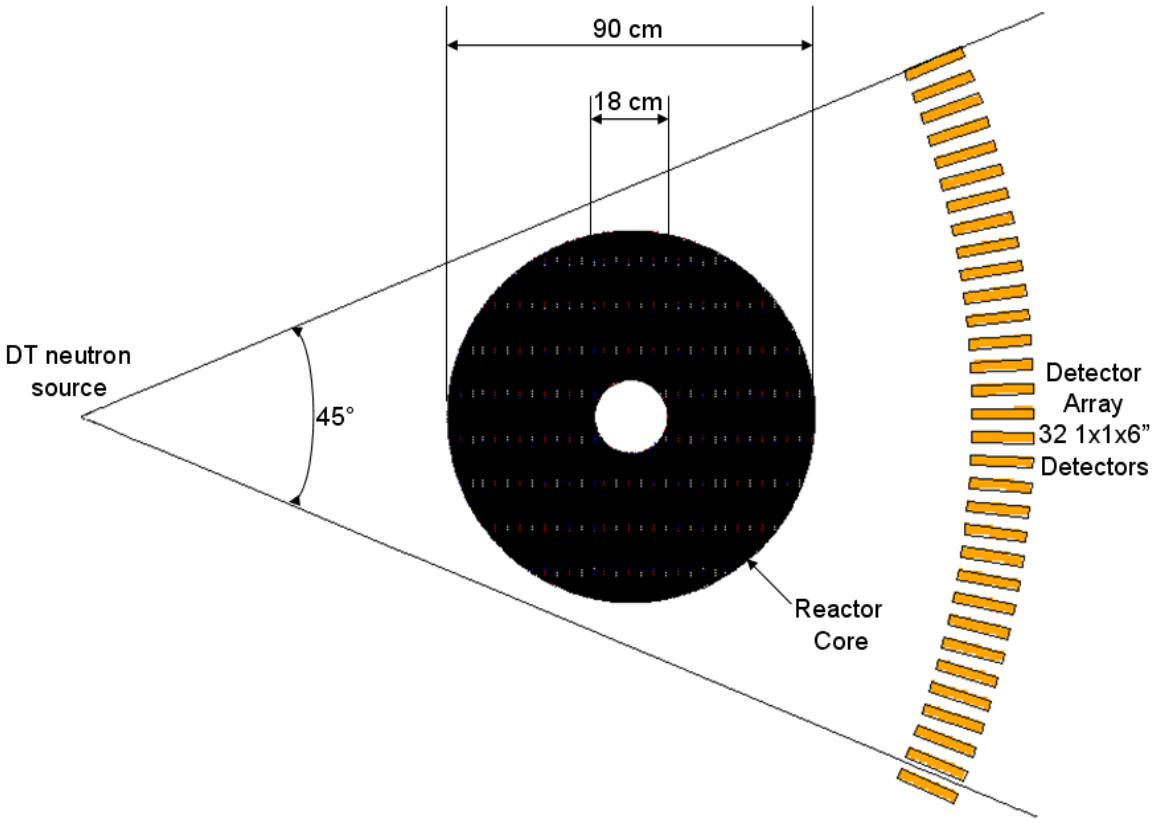

Figure 3. Horizontal cross-section of a reactor core being scanned by the NMIS array. The core shown here is the DTo3 (See Table 2) configuration. DT generator and electronics are omitted for clarity.

Figure 4 shows a plot of attenuation versus angle for all of the cases in the first scenario. All of the attenuations appear to differ from the baseline (no void) case, but the significance of the difference needs to be quantified.

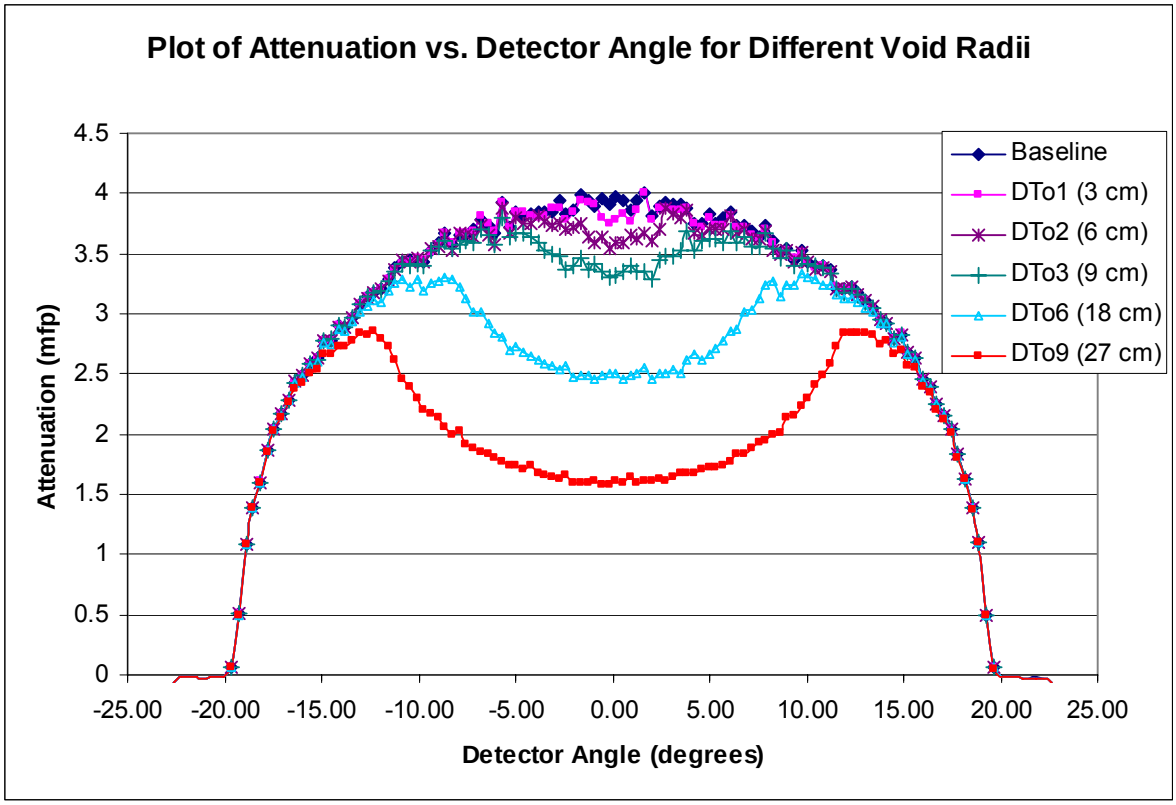

Figure 4. A plot of attenuation versus detector angle for the simulations in scenario 1. (Void radius is in parentheses in legend).

Figure 5 shows a close-up of the center 10° of the attenuation plots. In this plot, error bars showing 2.575 standard deviations represent a 99% probability that the true attenuation value lies within the error bars. For the purposes of this paper, two points are considered to be statistically different with a high (>99%) confidence level if their respective error bars do not overlap. In Figure 5, scenarios with a void radius of 6 cm or larger produce an attenuation plot statistically different from the attenuation of a core with no fuel removed. The attenuation of scenario DTo1 (3 cm void) is not statistically different (the error bars of this curve were omitted for clarity.) Thus the threshold of detectibility in this scenario is between 0.67 and 2.67 kg of ^{235}U removed. This amount represents approximately 1% of the total fuel inventory.

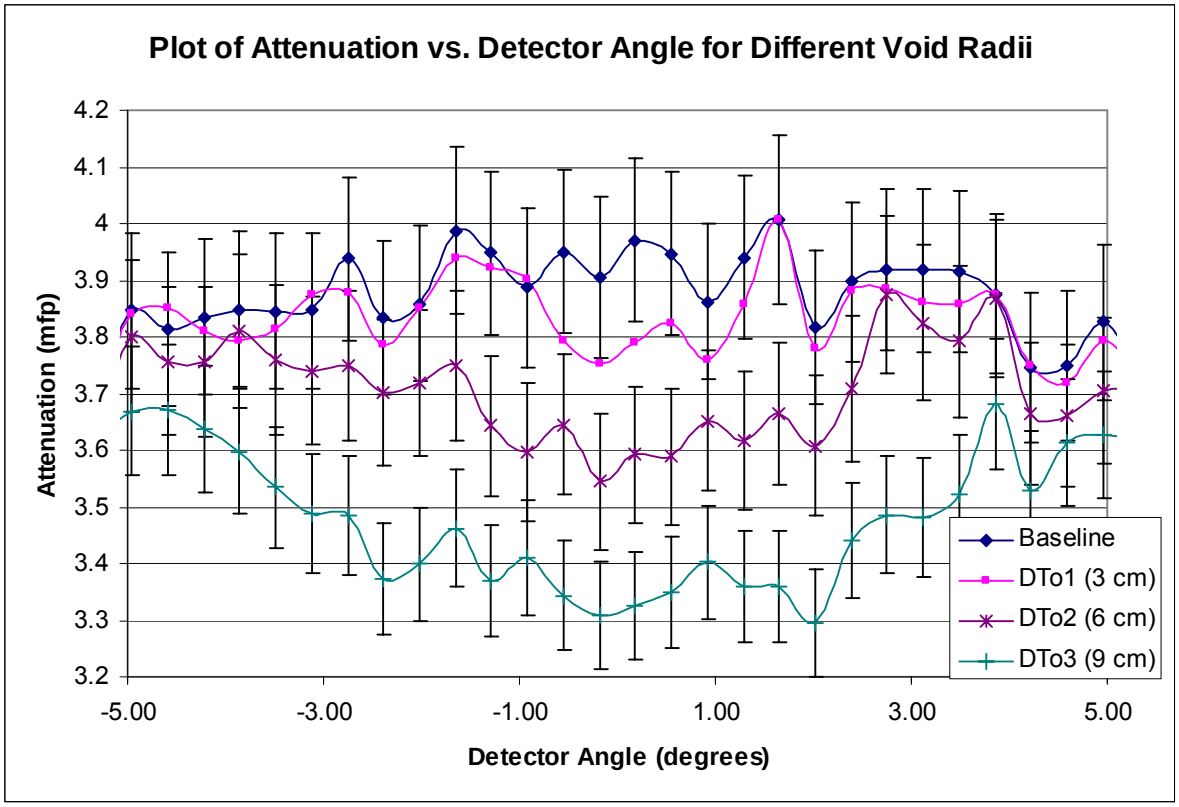

Figure 5. A close-up plot of attenuation versus detector angle for the simulations in scenario 1. Error bars represent 99% confidence intervals. Error bars for the DTo1 simulation overlap the baseline but are omitted so that the gap between the DTo2 and baseline error bars can be seen.

5.2. Scenario 2

In the second series of simulations, fuel pins were removed symmetrically throughout the reactor. This series of simulations was designed to show that NMIS with imaging can detect the uniform removal of fuel pins in addition to the bulk removals of scenario 1. This was accomplished in the model by defining a notional assembly consisting of 27 fuel pins and removing 1, 2, or 3 pins from each assembly. Table 3 shows the details of the three models. These models were compared with the DTo0 case (no fuel removed) from the first series of models to see the effect of the missing fuel pins on the attenuation map. Figure 6 shows a close-up view of the one of the reactor assemblies with 1 fuel pin per assembly removed.

Title	Missing pins / Assembly	Void Fraction (%)	Missing Mass ^{235}U (kg)
Baseline	0	0	0
1gReg	1	3.70	5.56
2gReg	2	7.41	11.11
3gReg	3	11.11	16.67

Table 3. Parameters used for simulations in scenario 2.

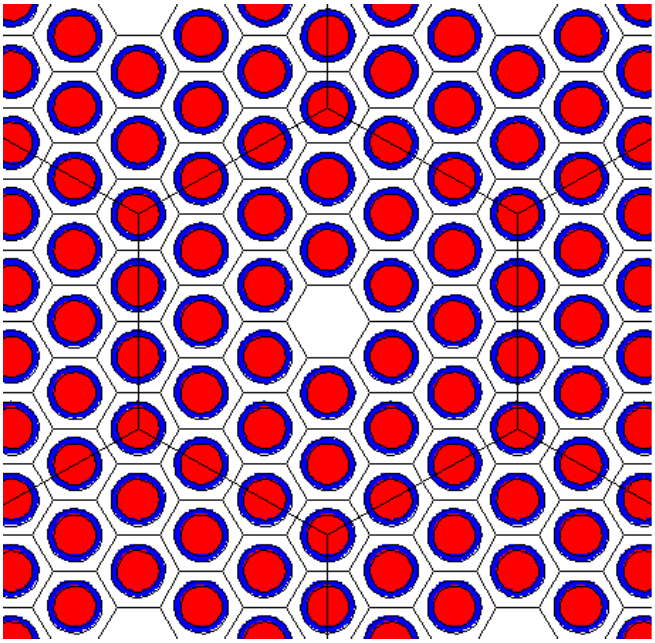

Figure 6. A cross-section of a notional hexagonal array with one fuel pin removed from the center. Note that hexagonal structures are for modeling purposes only and do not represent physical structures inside the reactor core.

Figure 7 shows the attenuation plots for the second scenario, the removal of 1, 2, or 3 fuel pins from each notional array in the reactor.

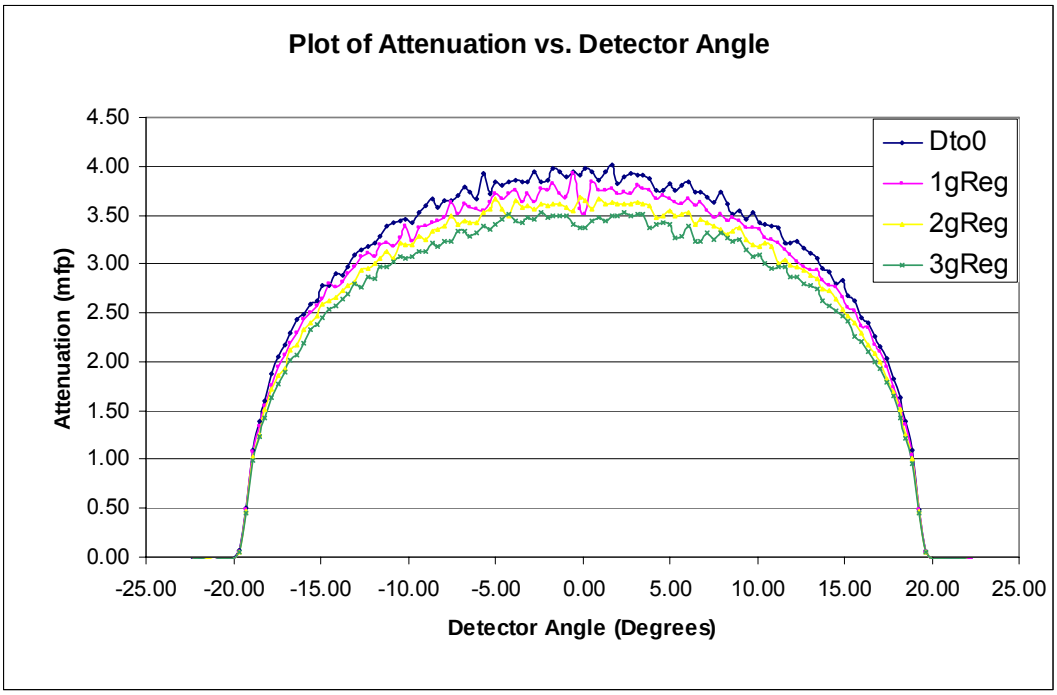

Figure 7. A plot of attenuation versus detector angle for the simulations in scenario 2.

A close examination (not shown) of the attenuation plots from Figure 7 would seem to indicate that, with the exception of a small dip in the center created by the alignment of the fuel pin voids, the 1gReg case (1 fuel pin removed per assembly) is not statistically different from the DTo0 case (no missing fuel). This is because the reduction in attenuation due to the removal of fuel pins is spread across all

detector angles. For this scenario, a more appropriate analysis technique involves fitting the entire attenuation curve and comparing the uncertainty of the fit with that of the baseline case. By comparing the entire attenuation curves in this manner, effects which are not visible on a point by point basis can be detected. For a cylindrical reactor core, an excellent fit of the attenuation can be made using the equation

$$e^{\tau} = a + b\theta^2 + c\theta^4 \qquad \text{(Equation 3)}$$

where τ is the attenuation in mean free paths and θ is the detector angle in degrees. The points were weighted by the number of counts in the non-void simulation. Prior to the fit, all points outside the limits of the reactor core were removed. The JMP 7 [14] statistical software package was used to calculate the best values of the fit parameters. Table 4 shows the final values of the fit parameters for the three cases in this scenario plus the baseline case.

Title	a	b	c (× 10^{-4})
Baseline	49.92 ± 0.32	-0.1933 ± 0.0035	1.73 ± 0.080
1g	42.21 ± 0.34	-0.1542 ± 0.0038	1.23 ± 0.087
2g	37.33 ± 0.20	-0.1338 ± 0.0023	1.03 ± 0.053
3g	32.29 ± 0.20	-0.1103 ± 0.0023	0.76 ± 0.053

Table 4. The final fit parameters for the simulations in scenario 2.

Figure 8 shows the central region of the fitted attenuation curves. The error bars still represent a 2.575 σ (99%) confidence level, but in this case they represent the uncertainty of the fitted curve rather than the individual data points. This uncertainty was computed by propagating the uncertainties shown in Table 4 using a standard technique for propagation of uncertainty. As Figure 8 shows, using this analysis method the attenuation of the 1gReg case can easily be differentiated from the baseline case with no voids. The size of the gap between the two cases suggests that the limit of detectibility in this scenario is much less than the 5.56 kg (3.7%) removed in the 1gReg case.

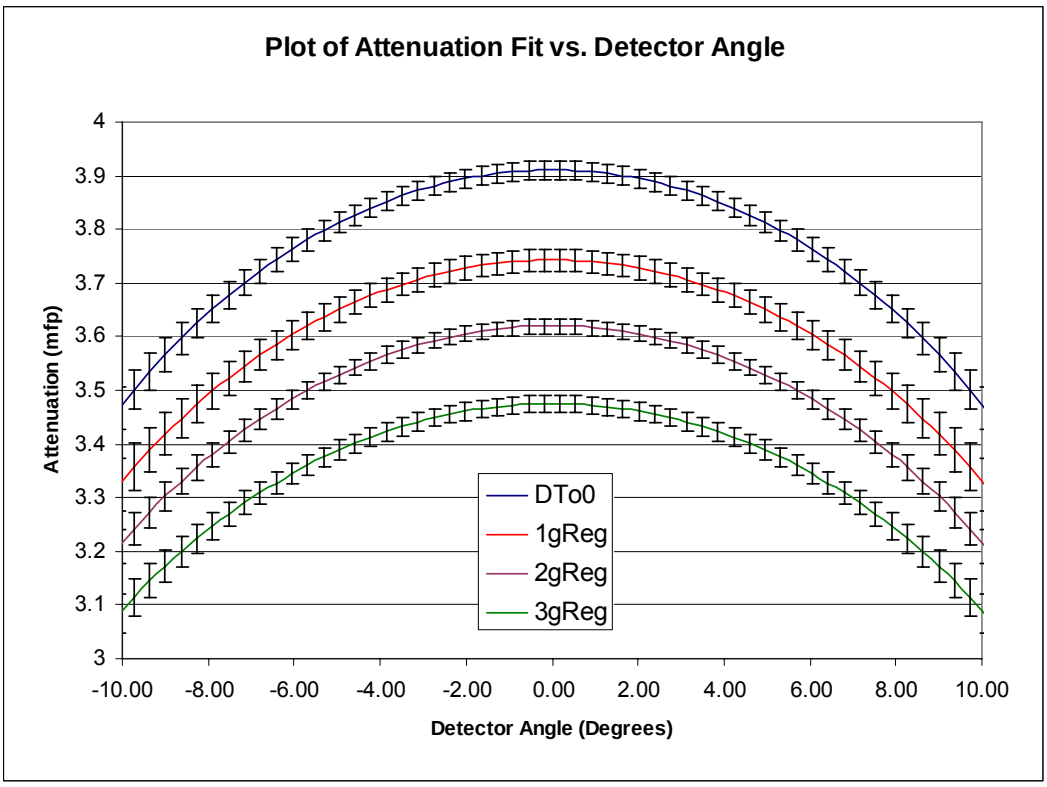

Figure 8. A close-up of the fit attenuation curves for the simulations in scenario 2. The error bars represent a 99% confidence interval of the fit.

5.3. Scenario 3

A final series of simulations repeated the second series but increased the density of the remaining fuel pins so that the total fuel inventory remained the same as the baseline despite the presence of the voids. This series of simulations was performed to show that the presence of small inhomogeneities in the reactor core would not alter the results so long as the total amount and composition of material between the source and each detector remained constant. Table 5 shows the details of these simulations.

Title	Missing pins/ Assembly	Void Fraction (%)	Missing Mass ^{235}U (kg)
Baseline	0	0	0
1gEnh	1	3.70	0
2gEnh	2	7.41	0
3gEnh	3	11.11	0

Table 5. Parameters used for simulations in scenario 3.

Figure 9 shows the results of the scenario 3 simulations. With the exception of the small dip in the center of the 1gEnh case due to the alignment of the voids, there is no discernible difference among the four attenuation curves. Indeed, a fit of the four scenarios (not shown), such as the one from Scenario 2, shows that none of the curves is statistically different from the baseline case at the 99% confidence level. This indicates that the NMIS imaging is not particularly sensitive to the small-scale (fuel pin size) structure within the reactor core, but rather to large-scale (assembly size) structures such as large voids and the total amount of fuel in the core.

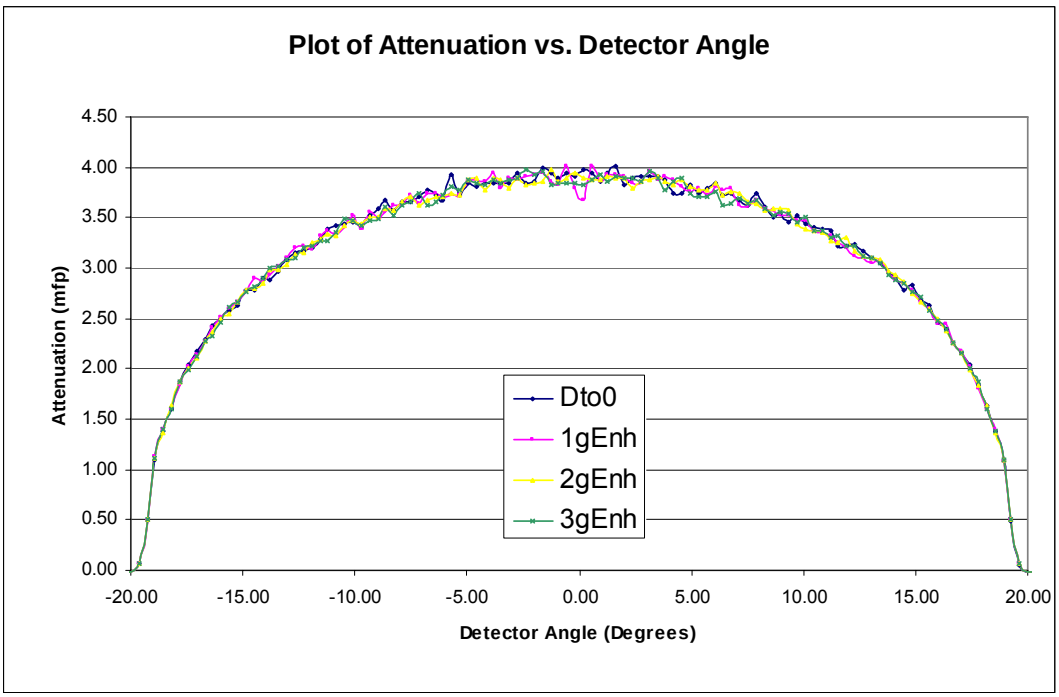

Figure 9. Plot of attenuation versus detector angle for the simulations in scenario 3.

6. Conclusions

As the demand for energy increases, nuclear energy is expected to account for a significant portion of new power plants. In developing countries and for nonelectrical applications, SMRs should play a key role. NMIS with imaging provides a useful tool that inspectors can use for nonintrusive verification that the stated mass and enrichment of uranium is present in the completed reactor core at the output of the core fabrication facility. In conjunction with a validation of the containment, this method can be used for the initial certification of the completed reactor core.

These preliminary simulations have shown that NMIS can detect the removal of as little as 1% of the fuel inventory from a single location or the removal of 3.7% (and probably much less) throughout the reactor core using a 40 minute measurement. Future improvements in the DT generator source strength can reduce the measurement time by a factor of 3 or more. The simulations also showed that the presence of voids alone did not significantly affect the attenuation plot so long as the total amount and composition of material between the source and each detector remained approximately constant. This would allow for verification of the fuel inventory while revealing considerably less information about the small-scale design of the reactor.

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8. Disclaimer

SESSION 12

EXPORT CONTROL

Euratom Co-operation Agreements, Safeguards, and Export Controls

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Abstract:
UNSCR 1540, inter alia, requires all states to establish domestic export control laws to prevent proliferation of chemical, biological and nuclear weapons. Most nuclear supplier states apply the NSG's Guidelines when making nuclear transfers, as indeed do all the member states of the European Union.

The Euratom Community in the 50 years of its existence has signed a number of international agreements with major trading partners concerning the supply of nuclear materials or equipments. These agreements usually include provisions on Safeguards, or provisions implementing one or both parties' export control policies, thereby conferring operational responsibilities upon Euratom as a party to an agreement.

In view of the ongoing nuclear renaissance and the universalisation of export controls by UNSCR 1540, it is likely that the number of Euratom Supply Agreements will continue to increase over the next few years.

This paper presents an inventory of the currently in force Euratom Co-operation Agreements, coupled with analysis of their substantial content, placing the emphasis on the elements relating to safeguards or parties' export control policies.

This paper also examines the Commission's practical role in the execution of these agreements, and raises some areas where the Commission may experience practical difficulties in executing its obligations under the agreements.

Keywords: Export Controls, Euratom Agreements,

1. The Euratom Treaty

The Euratom Treaty was signed in 1957 by the representatives of Belgium, France, Germany, Italy, Luxembourg, and the Netherlands. Since then, successive enlargements have brought the number of Euratom member states to 27.

Article 1 of the Treaty (reproduced below) sets out the Euratom Community's overarching task.

It shall be the task of the Community to contribute to the raising of the standard of living in the Member States and to the development of relations with the other countries by creating the conditions necessary for the speedy establishment and growth of nuclear industries.

Article 2 (reproduced below) sketches out this task in more detail.

In order to perform its task, the Community shall, as provided in this Treaty:

a. *promote research and ensure the dissemination of technical information;*

b. *establish uniform safety standards to protect the health of workers and of the general public and ensure that they are applied;*

c. *facilitate investment and ensure, particularly by encouraging ventures on the part of undertakings, the establishment of the basic installations necessary for the development of nuclear energy in the Community;*

d. *ensure that all users in the Community receive a regular and equitable supply of ores and nuclear fuels;*

e. *make certain, by appropriate supervision, that nuclear materials are not diverted to purposes other than those for which they are intended;*

f. *exercise the right of ownership conferred upon it with respect to special fissile materials;*

g. *ensure wide commercial outlets and access to the best technical facilities by the creation of a common market in specialized materials and equipment, by the free movement of capital for investment in the field of nuclear energy and by freedom of employment for specialists within the Community;*

h. *establish with other countries and international organizations such relations as will foster progress in the peaceful uses of nuclear energy.*

The detailed substantial provisions of the Treaty are set out in the ten chapters of its second Title. As far as this paper is concerned, the most relevant chapters are Chapter 6 (Supplies), Chapter 7 (Safeguards), and most particularly Chapter 10 (External Relations).

The Euratom Treaty entered into force in 1958. Evidently much of the period immediately following entry into force was dedicated to making the Euratom Commission operational, but the first Euratom legislation had already been adopted by the end of 1958 [1], including the Chapter 2 security rules [2] and the Euratom Supply Agency's statutes [3]. During 1959 rapid progress continued to be made, including the adoption of the first Regulations on Euratom Safeguards [4, 5, 6]. As regards the subject matter of this paper, the first nuclear co-operation agreements between Euratom and third states were signed in 1959 [7, 8].

2. Euratom External Relations

In the late 1950's the then member states of the Euratom Community possessed little civil nuclear expertise, and had access to only limited quantities of ores and source materials. These gaps could only be filled through co-operation with friendly states (since the Cold War was near its peak) possessing nuclear know-how and having access to nuclear materials. The domain being so sensitive, international nuclear co-operation was, and still is, normally conditional upon the existence of a state to state agreement under international law.

However, even friendly states did not wish to see the emergence of further states with nuclear weapons. Access to nuclear materials for civilian purposes was made conditional upon the provision of guarantees (i.e. safeguards) that supplied materials would not be diverted to assist somehow in the fabrication of nuclear weapons. IAEA safeguards, not being operational at the end of the 1950's the choice was between safeguarding by the supplier states, or some form of supranational control. In the event, the latter prevailed, resulting in Euratom Chapter VII safeguards.

It was also necessary for the young Community to ensure that all users within the Community would benefit from equitable access to nuclear materials for their civil nuclear programmes – hence the Treaty's Chapter VI on supplies, which also established the Euratom Supply Agency.

Returning to the Community's external relations, as noted above the most important element would be its international agreements, and particularly those including supply of materials and equipment. The legal basis for these agreements is Article 101, which empowers the Community – where it has jurisdiction – to enter into agreements with third states and international organisations. Agreements are made between the Community and the third party. The Commission represents the Community, but is not a party to the agreement. The Commission is also responsible for negotiation of agreements, but nonetheless acting under a mandate from another institution, the Council, representing the member states of Euratom. Article 184 confers legal personality upon the Euratom Community.

3. Nature of agreements

This paper is about those Euratom Agreements, the scope of which includes provisions on commerce in nuclear items and materials. The agreements usually also deal with more general co-operation, such as in research, or in combating illicit trafficking. Indeed, only one Euratom supply agreement deals purely with nuclear supply.

The Euratom Community's competences are limited to the peaceful uses of nuclear energy,

consequently the scope of Euratom agreements is equally limited to peaceful co-operation.

The Euratom co-operation/supply agreements discussed in this paper are bilateral, that is to say that the parties are the Community and a third state. (Agreements involving the Community, a third party and one or more member states are also possible, for instance the safeguards agreements with the IAEA).

Note also that nuclear commerce with third states is not an exclusive competence of the Community; member states may also sign agreements with third states provided that their agreements do not impede the operation of the Euratom Treaty or its derived legislation. (Article 103 sets out a procedure for the Commission to verify members states agreements with third states or international organisations whilst they are still at the draft stage).

Trade in nuclear materials and equipment is sensitive and therefore normally conditional upon the existence of an agreement at state to state level. Although at present, there is no Euratom nuclear co-operation agreement with Russia, several EU member states engage in nuclear trade with Russia under bilateral agreements with Russia. This is not prohibited under the Euratom system, provided the agreements are consistent with the Treaty. The question of whether or not to put in place a Euratom agreement as opposed to one or more national agreements is more practical than legal.

The co-operation agreements set trade in a framework of requirements related to the use made of the supplies and to non-proliferation requirements more generally. These requirements are usually known as "obligations", and the agreements normally provide for some form of demonstration of compliance with these "obligations". The Commission, in its capacity as representative of the Community, is responsible for supplying partners with these demonstrations of compliance. To a large extent, they concern accountancy and control. The regulation on Euratom safeguards does not include any provisions specifically tailored to gathering information in this connection, and nor do the recitals to the regulation mention the Euratom agreements, even though Article 77b of the Treaty specifically makes mention of obligations arising from international agreements. However, the safeguards regulation does include provision for reporting of advance notification of imports, even though the primary reason is to meet the Community's other safeguards commitments.

Currently, seven Euratom Agreements which include supply provisions are in force. Table 1 below summarises their dates of entry into force and their durations. Table 2 indicates whether the agreements' provisions apply equally in both directions, as well as their technical scope.

Euratom agreements represent the result of a negotiation, meaning that however convenient it would be to have a standard Euratom agreement, in practice this cannot be achieved. Tables 1 and 2 below indicate some of the variety to be found in the agreements. The variety can be seen in the duration of agreements, and in their technical scope. As will be seen later, there may also be some variation in the safeguarding and non-proliferation obligations resulting from the agreements.

"Equipment" normally means major systems such as complete reactors, or major items of plant such as steam generators or pumps for the primary coolant circuit. However, equipment may also extend down to the level of zirconium cladding for fuel pins. Agreements may also include provision for the parties to mutually designate items of equipment in order to effect transfers under the agreement. "Non-nuclear materials" means heavy water, deuterium, or nuclear grade graphite.

Table 1. Status of Currently in Force Euratom Supply Agreements

Partner	Entry into force	duration	comments
USA	1995	30 years minimum	(renewal of the 1959 agreement)
Canada	1959	Not defined	
Australia	1981	2012	
Japan	2006	30 years minimum	
Ukraine	2006	5 years renewable	
Kazakhstan	2009	5 years renewable	
Uzbekistan	2004	5 years renewable	

Table 2 – Nature & Technical Scope of Areements

Partner	Reciprocal obligations ?	Scope	comments
USA	yes	nuclear and non-nuclear materials, equipment	material tracking
Canada	yes	nuclear and non-nuclear materials, equipment	material tracking
Australia	no	nuclear materials,	material tracking, supply by Aus only
Japan	yes	nuclear and non-nuclear materials, equipment	material tracking (specific)
Ukraine	yes	nuclear materials	no material tracking
Kazakhstan	yes	nuclear materials	no material tracking
Uzbekistan	yes	nuclear materials	no material tracking

4. Nuclear Export Controls

During the last 50 years, an international framework has gradually been erected to ensure that the widespread adoption of nuclear energy is not accompanied by horizontal proliferation – that is a proliferation of the number of states possessing nuclear weapons.

The earliest mechanisms for the provision of verifiable guarantees of non-proliferation came at the end of the 1950's with the setting up of Euratom and IAEA safeguards. A decade later came the grand bargain of the NPT which would allow states which renounced nuclear weapons access to peaceful nuclear energy in return for accepting NPT safeguards – also known as comprehensive safeguards.

The NPT also created a basis in international law for export controls, through its Article 3.2 which provides that

> *Each State Party to the Treaty undertakes not to provide: (a) source or special fissionable material, or (b)* ***equipment or material especially designed or prepared*** *for the processing, use or production of special fissionable material, to any non-nuclear-weapon State for peaceful purposes, unless the source or special fissionable material shall be subject to the safeguards required by this Article.*

The NPT contains no technical annexes. The first common understanding of what constitutes **'equipment or material especially designed or prepared'** was prepared in the early 1970's by a group known as the Zangger Committee. This list came to be known as the Trigger List.

Nowadays, Zangger has been eclipsed by an international group of states, known as the Nuclear Suppliers Group, which draws up guidelines on nuclear transfers. The current version of the NSG's Guidelines dates from November 2007 [9], and includes the NSG's version of the Trigger List, which is almost identical to Zangger's Trigger List. The NSG Guidelines are an international gentlemen's agreement and are politically, not legally, binding. There are currently 45 members of the Nuclear Suppliers Group, including all the major suppliers of nuclear materials or equipment other than Niger and Uzbekistan. All the EU member states are members too, since by virtue of the single market they are all potential nuclear suppliers.

The European Commission has the status of Permanent Observer in the NSG, because of the Community's role in international nuclear trade resulting from the Euratom supply Agreements, because of the need to protect the Single Market against restrictions which might be drawn up in the framework of the NSG, and because of the Common Commercial Policy of the European Community Treaty (Article 133), which governs external trade in general.

The most important recent development in nuclear export controls was the adoption of Resolution n° 1540 by the United Nations Security Council in April 2004. This Resolution is binding upon all UN member states in accordance with Chapter VII of the UN Charter. It obliges states to develop and enforce legal and regulatory measures against the proliferation of chemical, nuclear, and biological weapons of mass destruction, related items and their delivery systems.

The adoption of UNSCR 1540 was one of the chief reasons for the EU to revise its own legislation on exports of Dual-Use items: the revised Regulation was adopted early May 2009 [10]. The significance of UNSCR 1540, in terms of Euratom Agreements, is that in future one can expect a greater stress to be laid upon the export control related aspects of the agreements.

5. Agreements and Export Controls

The obligations contained in the Euratom supply agreements represent export controls. Nonetheless, insofar as exports from the EU are concerned, the decision upon whether or not to authorise an export is taken by the authorities of the concerned member state, in accordance with the Community's Dual-Use Regulation, and not by the Commission as executor of the agreement.

Euratom supply agreements contain clauses covering some or all of the following safeguarding obligations :

advance notification of transfers
tracking
derived nuclear material
provisions on re-export,
peaceful use,
international safeguards, including fallback
further processing
physical protection

These obligations are discussed in turn below.

The purpose of providing advance notification of exports is twofold: first, to obtain assurances from the authorities in the destination that the receiver is duly authorised to receive the materials, and secondly in order to obtain confirmation that the transfer will indeed take place under the terms of the agreement.

The procedures governing advanced notification vary between the agreements, but one can distinguish two types of procedure – notification with confirmation of delivery, and notification without confirmation.

The chief means used to track nuclear materials is obligation accountancy. Material coming into the Community, which is subject to some or all of the obligations listed above, is assigned an obligation code (or more colloquially a "flag"). This sort of accountancy exists in the Euratom Safeguards System, but not in the IAEA Safeguards System, and thus the "flags" associated with materials are not reported by Euratom to the IAEA.

Euratom Safeguards also verifies the correctness of European operators' obligation accountancy. Indeed, in the absence of this sort of obligation accountancy and control it is difficult to see how states can provide assurances regarding the respect of safeguarding obligations on imported materials.

The agreements with Canada, USA, and Japan require the preparation of annual reports by both parties on the situation of material under their jurisdiction carrying the other party's safeguarding obligations. However, the Australian agreement, not being reciprocal and not involving transfers to Australia, only requires annual reporting by Euratom.

In some cases, nuclear material can become the subject of obligations not only upon import from the state of origin, but also as a result of being processed in a third state, or even in the destination state as a result of production or contact with equipment from another state which itself is subject to safeguarding obligations. Materials which acquire an obligation in this fashion are known as "derived" materials. Sometimes the terms "contamination" or "colouring" may be used to describe the production of derived material. A given batch of nuclear material can thus

acquire multiple safeguarding obligations if a series of such operations occur.

If this sounds complicated, it is because it really is complicated. A concrete, but realistic example, shows the sort of multiple "flags" which can arise in practice. Source materials from Australia are shipped to the USA for conversion and enrichment prior to shipment to Europe. Under the terms of the Euratom-Australia and Euratom-USA agreements the material would acquire both US and Australian "flags". Upon irradiation in a power reactor, not only is the material still wearing a US-Australian hat, but also the plutonium bred in the reactor is also wearing a US-Australian hat.

Now in respect of materials carrying multiple obligations, the uses to which the material can be put will be governed by the most restrictive of the safeguards obligations which it is carrying. If new nuclear supply agreements were to add new safeguarding obligations then it is clear that things would rapidly become unmanageable. For example, the Euratom-US agreement even contains an article addressing the resolution of difficulties arising from the overlapping of multiple obligations.

Some of the Euratom Supply agreements have their origins in the late 1950's and even when agreements are renewed or new agreements are negotiated, there is an understandable tendency to use existing agreements as a basis for discussions. Nonetheless, nowadays there is an internationally accepted framework of conditions to be applied to nuclear trade – the NSG Guidelines. If all parties to nuclear trade agreements agreed on complete alignment with the NSG Guidelines then in principle, if not in practice, follow up of international safeguarding obligations would be greatly simplified. Exact alignment with the NSG Guidelines is required rather than application as a minimum standard.

However, members of the international community have diverse opinions on nuclear commerce, and even members of NSG may well insist on provisions in agreements which go beyond what the NSG Guidelines require: for instance in respect of nuclear commerce with the nuclear weapons states.

All the Euratom supply agreements recognise the Community as a single entity, and so no agreement imposes any controls on retransfers of supplied materials or equipment within the Community, (although for technical reasons the agreement with Japan requires notification of transfers involving the EU's two nuclear weapons states). However, retransfers outside the EU or outside the territory of the partner are subject to conditions. The agreements with Kazakhstan, Uzbekistan, and Ukraine are fully aligned with the retransfer procedures set out in the NSG Guidelines, whereas the other four agreements contain specific procedures involving written prior consent except in the case of transfers to states on the partner's "White List" of states benefiting from generic prior consent. Euratom's own "White Lists" are drawn up in consultation with the member states.

A second type of prior consent requires the uses to which "flagged" material can be put within the Community. Two of the current supply agreements require prior consent for enrichment of uranium beyond 20% or for reprocessing. In fact, one agreement prohibits enrichment of supplied uranium beyond 20%.

All the agreements require that supplies be used for peaceful purposes only. Insofar as nuclear materials are concerned, and in the absence of other more specific safeguarding obligations, within the EU a special flag is applied in the accountancy to indicate peaceful use only for the materials.

All the agreements impose IAEA safeguards on both parties, and in the case of the Community, also Euratom safeguards. And all the agreements, with the exception of Canada, also make provision for the application of fall-back safeguards should it prove impossible to apply IAEA safeguards. For the EU NWS the concept of voluntary offer of facilities eligible for IAEA safeguards is applied.

The Euratom agreements require the application of physical protection measures in line with those set out in the Convention on the Physical Protection of Nuclear Materials (CPPNM). In addition, all save Canada – the oldest agreement, also require reference to the IAEA's guidance on physical protection set out in INFCIRC/225. The reference to this Guidance and to the CPPNM is standard practice on the international level, but given that both originated in the 1970's one has to wonder whether the adoption of a new international standard in this connection is overdue.

6. Agreements and internal market

The question also arises as to whether the supply agreements impose restrictions on transfers between member states of the European Community. All the supply agreements respect the logic of the Single Market. No agreement imposes restrictions such as a requirement to obtain prior consent for transfers between member states of the Community. It is also worth noting that this freedom of movement extends even to those items generally considered to be the most sensitive, such as highly enriched uranium. This indicates the confidence of the Community's partners in the Community's commitment to non-proliferation.

Some supply agreements require tracking of material or equipment covered by the agreement whilst the material or equipment is within the Community. However, this does not constitute a limitation on transfer.

Again, respecting the logic of the Single Market the agreements make no provision for exchange of government to government assurances in respect of transfers between member states, although some MS seem to still practice such exchanges.

7. Miscellaneous matters

Since the Euratom Treaty was signed by the six original member states in 1959, the Community has been enlarged six times bringing the total number of member states to 27. Many of the new member states have or had nuclear programmes of their own, including bilateral nuclear co-operation agreements with other states. If any of these agreements overlap with Euratom nuclear co-operation agreements then the new member states are obliged to enter into negotiations with the third state in view of the Community taking over as far as possible the overlapping obligations (Euratom Treaty, Article 106). In practice, these bilateral agreements tend not to have exactly the same scope as Euratom co-operation agreements, and therefore the negotiations concern suspension of those provisions of the state to state agreement which overlap with a Community agreement.

All the Euratom supply agreements include a requirement for the parties to negotiate operational procedures. Indeed, the agreements do not become operational in the absence of operational procedures, specifying at the working level formats, deadlines to respect, procedures for making notifications and so forth. Such operational procedures are in place for all the Euratom supply agreements, with the exception of the agreement with Uzbekistan.

As noted previously, some agreements require tracking of nuclear materials, or non-nuclear materials, or of plant and equipment. The legal framework of Euratom safeguards provides the Commission with all the tools it needs in order to be able to track and account for nuclear materials subject to safeguarding obligations under a supply agreement.

This legal framework, however, does not provide the Commission with the tools it would need to track non-nuclear materials or plant and equipment. In practice though, most of the European nuclear programme now in place was developed within the EU, and hence not subject to any obligations under Euratom agreements. As for externally supplied equipment, most of it was not supplied under a Euratom supply agreement. The remainder, externally supplied non-nuclear materials or equipment, supplied under the terms of a Euratom agreement, is sufficiently limited that the Commission can track it by informal contacts with installations. Therefore, the Commission sees no need at present to provide it with additional legal tools in this respect.

8. Added Value of Agreements

As noted previously, member states may also maintain bilateral nuclear co-operation agreements, with suppliers provided those agreements do not interfere with the Euratom Treaty system. Seen from the supplier's perspective however, it is evidently more attractive to negotiate a single agreement covering a group of states and having harmonised provisions and procedures in respect of all those states, rather than negotiating and maintaining a series of bilateral agreements with those states. This preference for Community level agreements can only be strengthened by the fact that more and more EU member states are discovering, or rediscovering, nuclear power.

A second aspect is connected to the need to secure access to supplies of nuclear materials and to ensure that all the member states have equitable access to the supplies.

The agreements contribute thus to the creation of a predictable level playing field. This is further encouraged by recent efforts of the EC and its three main partners (USA, Canada and Australia) to reach common understandings and to align the implementation practices of all their respective cooperation agreements .

9. Summary and Conclusions

The Euratom nuclear co-operation agreements represent the European Union's earliest international agreements. One of these agreements (Canada) celebrates its 50th birthday in 2009. , The Commission is central to the operation of the supply agreements, and has been so for 50 years.

At present, the Commission is able to meet its obligations on tracking of non-nuclear materials and equipment, through voluntary co-operation with installations, and therefore sees no need to reinforce its legal tools.

This paper has also explored some of the practical difficulties involved in the operation of the agreements, most particularly whenever nuclear material acquires safeguarding obligations from more than one state. International alignment with a harmonised set of safeguarding obligations such as those published in the NSG Guidelines would help eliminate these difficulties. However, for a number of reasons this is unlikely to happen in practice.

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Export Controls and the Model Additional Protocol

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Abstract:

Nuclear safeguards and export controls have been mutually dependent since Article III.2 of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) explicitly connected these two pillars of nuclear nonproliferation. As safeguards were strengthened with the Model Additional Protocol (INFCIRC/540) and as export controls were strengthened with the development of the Nuclear Suppliers Group (NSG) Dual- Use Guidelines (INFCIRC/254/Part 2), the relationship between export controls and safeguards have become more subtle and complex. Beyond the obvious connections between the Additional Protocol's (AP) Annex II and the NSG's "Trigger List" and the implicit dependence of the AP's export reporting requirements upon a functioning export control system; many other aspects of AP implementation depend on export control processes and understanding. This paper will explore the unstated connections between export controls and the AP's Annex I activities functionally related to the nuclear fuel cycle. This paper will ultimately argue that as safeguards are called upon to not only verify the correctness of state declarations but also their completeness, with the associated shift in focus towards discovering undeclared activities, it is essential to bridge the traditionally distinct disciplines of safeguards and export controls.

Keywords: additional protocol; export control; dual use; safeguards; training

ORIGINS

For those states seeking to fully and effectively implement strengthened safeguards and strengthened export control measures there are opportunities to take advantage of their many mutually reinforcing synergies. Correlations between nuclear safeguards and export controls were identified in a joint study conducted for the Department of Energy National Nuclear Security Administration Office of International Regimes and Agreements. The study states that "the true picture of a country's intentions, with regard to nuclear weapons, can be derived only by an integration of all components of the international nuclear nonproliferation regime, and in particular export controls and safeguards."[ii] This study's conclusions are timely in light of current proliferation threats and the anticipated nuclear renaissance, which will create challenges for existing nuclear safeguards and export control systems.

Article III of the 1970 the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) requires each State Party to the Treaty to undertake the implementation of safeguards and to limit nuclear trade to activities under the IAEA safeguards system. NPT Article III.2 states:

> Each State Party to the Treaty undertakes not to provide: (a) source or special fissionable material, or (b) equipment or material especially designed or prepared for the processing, use or production of special fissionable material, to any non-nuclear-weapon State for peaceful purposes, *unless the source or special fissionable material shall be subject to the safeguards* required by this Article. [iii] [Emphasis added]

The first parallel steps in export control and safeguards ensued during the early 1970's following the entry into force of the NPT with the development of Comprehensive Safeguards Agreements (CSA) and the formation of the NPT Exporters' (or Zangger) Committee to interpret the meaning of the phrase, "especially designed or prepared." The Zangger Committee's guidelines stipulate that a supplier's export of items on an accompanying list trigger the requirement for safeguards. These guidelines and the accompanying "Trigger List" were communicated to Member States by the IAEA though INFCIRC/209 in 1974. Supplier states then formed the Nuclear Suppliers Group (NSG) which issued additional voluntary Guidelines in 1978 (INFCIRC/254). At the same time, under NPT provisions, all non-nuclear weapon States were obligated to conclude a CSA with the IAEA within 18 months of becoming party to the Treaty.

Several developments in the 1980s and 1990s revealed weaknesses in both the export control arrangements and safeguards systems with respect to combating clandestine nuclear programs and verifying the completeness of State declarations to the IAEA. Following the 1991 Gulf War, IAEA inspections revealed that Iraq was developing a clandestine nuclear weapons program involving undeclared enrichment activities. While these activities occurred adjacent to inspected facilities, the IAEA lacked the authority to request access to those facilities. Also in 1991, the South African Government revealed that it had disarmed and dismantled its secret nuclear weapons program. Both cases underscored the dangers of unregulated dual-use transfers and related clandestine facilities. Largely in response to these developments, a flurry of activity ensued to strengthen both the safeguards system and the export control regime.

In 1993, in response to these and other events, the IAEA initiated "Programme 93+2" (reflecting the expectation of completion within two years) to expand safeguards to verification of not only the *correctness* of declarations, but also their *completeness*. The necessary additional measures fell into two categories: those covered by existing legal authority which had not been fully utilized and those requiring additional legal authority. For the former, the IAEA relies on a country's State System of Accounting and Control (SSAC) to provide all the relevant information required under its CSA.

In the area of new legal authorities, Programme 93+2 resulted in the 1997 promulgation of the "Model Additional Protocol" (INFCIRC/540). Upon voluntary adoption by Member States, the "AP" supplements existing safeguards agreements by providing the IAEA with additional authority for both information and physical access to draw conclusions about the non-diversion of material declared through traditional safeguards and to verify the absence of undeclared nuclear material or activities. Included in this extended scope of safeguards under an AP is broader IAEA access to sites and information related to nuclear fuel cycle research and development. The types of information that are reported to the IAEA under the AP include those research and development activities listed in Annex I of the Model AP, and information about the manufacture and export of sensitive nuclear-related items listed in Annex II of the Model AP. These are all traditional areas of competence for regulation by national systems of export control, which seek to ensure that sensitive nuclear-related items and information be exported only to those States with demonstrated commitments to non-proliferation.

However, the concurrent strengthening of international nonproliferation export control regime standards, while intended to address the same challenges as those addressed by the Model AP, proceeded independently. The key parallel development was the NSG's 1992 creation of additional (Part II) guidelines for transfers of nuclear-related dual-use equipment, material and technology which could make a significant contribution to an unsafeguarded nuclear fuel cycle or nuclear explosive activity.[iv] Many of the items on the Dual-Use List are manufacturing inputs which, not surprisingly, are used in AP Annex I activities functionally related to the nuclear fuel cycle.

While safeguards and export controls have always been linked, the need is now apparent to go further and integrate them in more practical ways, allowing for mutual reinforcement. As stated by IAEA Director General Dr. Mohamed ElBaradei, "…even a verification system making use of the authority under the Additional Protocol may not reliably detect low levels of clandestine nuclear activity, such as that conducted in Iran and Libya for many years, unless at the very least supported and supplemented by the sharing of actionable information from an effective system of export control…."[v]

LINKAGES

Potential synergies among the enhanced elements of safeguards and export control implementation at the national level include the dependence of the AP's export reporting requirements on a functioning national export control system and the necessity of government outreach to effected enterprises. Possible synergies at an international level include the potential requirement for an AP as a condition of nuclear supply and the potential for trade analysis to contribute to IAEA State evaluations. Finally, criticial to effective implementation of the safeguards and export control missions is the need for safeguards inspectors and export controllers to *both* understand the dual-use equipment and materials used in manufacturing activities functionally related to the nuclear fuel cycle. The remainder of this paper will explore each of these linkages.

Export Reporting Requirements

For exports of items listed in Annex II, the AP's Article 2.a.(ix) requires reporting to the Agency the identity, quantity, date of export, and location of intended use in the receiving State. The obvious similarity between AP Annex II and the Trigger List of the NSG's Part I Guidelines is no coincidence. Certain export control processes, such as export licensing and enterprise compliance, can facilitate correct and complete reporting. Hence, the governmental body responsible for nuclear export licensing decisions is the natural locus for the collection of such data at a national level before it is reported to the IAEA. If another agency has this responsibility, then interagency coordination is called for to ensure that its information is consistent with the national export control licensing agency's knowledge about those entities that possess or trade in Annex II-related commodities. Certain countries have in fact already identified and remedied inconsistencies between information collected by their licensing agencies and their agencies responsible for reporting Annex II exports to the IAEA.

Government Outreach

The organizations that are typically responsible for AP implementation are the regulatory bodies that control nuclear and radiological material and activities. These agencies normally have a good grasp on activities related to organizations with which they traditionally work, such as government-owned research institutions, nuclear enterprises, and universities. However, implementing the AP involves an expanded range of enterprises such as commercial industry and multinational firms engaged in Annex I manufacturing activities. It is typically the export regulator that has (or at least should have) a grasp on the activities of these entities. Since nonproliferation awareness and competency within these enterprises is an important part of ensuring export compliance and since the AP requires information from and preparations by these enterprises, government outreach by national regulatory and enforcement organizations presents an important area where safeguards and export control implementation can be mutually reinforcing. In fact, the implementation of an AP can serve as an especially important complement to export controls because export licensing agencies often have difficulty identifying entities involved in nuclear fuel-cycle manufacture and R&D that have not applied for export licenses. These entities – particularly small enterprises – are often the most vulnerable to inadvertently transferring nuclear-related commodities and knowledge. This implies the need for coordination between the involved licensing and IAEA-focused regulatory bodies, with the shared aim of identifying relevant enterprises for outreach and potential AP-related reporting, respectively.

AP as a Condition of Nuclear Supply

As stated above, safeguards have always been the principal conditions of nuclear supply, as articulated in the NPT's Article III.2. This condition has been refined over the years, including the NSG's requirement for so-called full-scope safeguards as a condition of supply for exports of items on the NSG's Trigger List. Now many suppliers within the NSG believe that this linkage needs to be extended to requiring an AP to be in force as a new condition of supply of enrichment or reprocessing technologies.[vi] While differences remain between responsible suppliers as to whether or not this condition of supply should be formally included in the NSG Part I Guidelines, it is generally acknowledged that trade with a recipient State that has an AP in force increases confidence among suppliers about that State's ability to effectively regulate and control nuclear-related holdings and transfers.

Trade Analysis

Declarations to the IAEA of manufacturing activities covered in AP Annex I as well as exports of especially designed or prepared nuclear equipment and material listed in Annex II provide an invaluable tool for the IAEA as the international agency responsible for ensuring the transparency of nuclear-related activities and for reporting to the international community. As expressed by Goorevich et al, "Export control information, especially with regard to attempted procurements of dual use items that could be serving proliferation-relevant activities, is potentially a significant contributor to the confidence of the IAEA in its safeguards conclusion for a state."[vii]

Dual-Use Equipment and Materials

While the relationship between the AP's Annex II and the NSG's Trigger List is apparent, there is a less visible relationship between the AP's Annex I and the NSG's Dual-Use List. AP Annex I concerns activities "functionally related to the nuclear fuel cycle", while the NSG's Dual Use List concerns "items that can make a major contribution to an unsafeguarded nuclear fuel cycle or nuclear explosive activity." While unstated, the overlapping intents create a relationship; NSG dual-use items are used in AP Annex I activities.

For example, AP Annex I.(i) deals with "the manufacture of centrifuge rotor tubes or the assembly of gas centrifuges." While complete gas centrifuges and rotor tubes are NSG Trigger List items, the materials and equipment used to fabricate centrifuge components are largely dual-use. These include high-strength metals (aluminum and titanium alloys and maraging steel), fibrous and filamentary materials (carbon, aramid, and glass fiber), flow forming machines, filament winding machines, and balancing machines, all of which are covered by the NSG's Dual-Use List. Similarly while AP Annex I. (xv) covers construction of hot cells, the NSG Dual-Use List includes radiation shielding windows and remote manipulators, which are the essential inputs for the construction of hot cells.

Two interesting implications emerge. First, with the AP giving the IAEA access to locations engaged in Annex I activities and authority to visually observe those locations, it becomes clear that IAEA inspectors need to become familiar with these activities and with the dual-use items used in those activities. Indeed, these items are the most visually identifiable indicators of Annex I activities, and as such are potential signal indicators of otherwise difficult to detect clandestine R&D. It should be noted that by their nature, dual-use goods are also (usually) used in non-nuclear activities. Dual-use goods by themselves are not necessarily indicators of a nuclear program, and producers of dual-use goods are not necessarily supporting or engaging in Annex I activities.

Second, from the export control perspective, not all of the activities listed in Annex I have corresponding dual-use control list entries on the NSG Dual-Use List, potentially suggesting some important areas related to the nuclear fuel cycle where the NSG list could add items so as to enable States to better coordinate export controls and safeguards under the AP.

Training for Inspectors

Understanding dual-use commodities associated with nuclear fuel cycle activities has traditionally been the purview of nuclear industry, governmental regulators, and national representatives to the WMD nonproliferation regimes like the NSG. However, as proliferation threats and the institutions intended to address them have evolved, it has become increasingly apparent that the capacity to visually identify both single use (Annex II and Trigger List) components and the dual-use components needed to manufacture them could be of use to two additional sets of officials. First, national export control enforcement officers need these skills and are in fact already being trained in many countries to recognize controlled items during the process of physical inspections of cargo at ports of entry and exit. Second, IAEA inspectors could use these same skills to confirm that State declarations are correct *and* complete during the course of inspections.

CONCLUSION

This paper calls attention to five synergies between AP and export control implementation. First, national export control systems should be leveraged to facilitate AP-related export reporting. Second,

coordinated government outreach by the nuclear and export regulatory bodies can benefit both AP and export control implementation. Third, requiring the AP as a condition of supply for enrichment and reprocessing technology would help address the concern of clandestine development of these sensitive nuclear technologies. Fourth, trade analysis can significantly contribute to the confidence of the IAEA in its safeguards conclusions. And fifth, the logical relationship between the AP Annex I activities and the NSG dual-use list implies that IAEA inspectors as well as national export control inspectors need to become more familiar with dual-use items than is currently the case.

United Nations Security Council Resolution 1540 calls for States to take action to strengthen both safeguards and export controls in the fight against proliferation of WMD. There are indeed compelling arguments to be made for safeguards and export controls to further the complementary and mutually-reinforcing strategies described in this paper, particularly for countries on the cusp of adopting nuclear power. To quote Goorevich et al, "…it is imperative to consider how to more effectively integrate export control and safeguards activities into single policy strands that are mutually supporting, to produce smarter, more efficient nonproliferation results." The nonproliferation benefits from an integrated approach to implementation of safeguards and export controls have already been realized in some quarters, but much more needs to be done to defeat illicit procurement activities capable of supplying clandestine facilities. Further integration at the national and international levels can help decision-makers make the wisest possible use of the limited resources and technical expertise needed sustain international nonproliferation norms.

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[i] The authors implement safeguards and export control outreach programs for the US Department of Energy National Nuclear Security Administration Office of Global Security Engagement and Cooperation.

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Exploring the use of trade data to support safeguards verification activities

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Abstract

As part of a European Commission Support Programme Task to the IAEA, a survey of world trade data from open source has been conducted. A catalogue was produced to cover services offering data on trade on all categories of goods, both of statistical and transactional nature. Services on statistical data have a worldwide coverage and are provided, often for free, by international and governmental organizations, and statistical offices. Services on trade transactions may have a national or multi-national scope and are typically provided by private companies. The survey was motivated by the hypothesis that trade databases may be useful to support IAEA verification activities. Trade data may provide insights to support the verification of additional protocol declarations and nuclear material transfers as well as give indications of possible undeclared activities. Since trade data is indexed in data services by the Harmonized System (HS) nomenclature of goods, a prerequisite to use this type of data is to 'map' items of interest to safeguards verification with HS categories. The paper reviews services available on world trade data with an emphasis on geographical and temporal coverage, type of data provided and accessibility. It then addresses the challenging task of correlating categories of items of interest and HS codes. The paper concludes on some possible uses of trade data for nuclear verification purposes.

Keywords: Safeguards, trade data, import/export, sensitive goods, additional protocol.

1. Introduction

Information on imports and exports to be declared to the IAEA under comprehensive safeguards agreements (CSA) [1] is limited to nuclear material and source material. Following the disclosure of undeclared nuclear programmes in Iraq, South Africa, and DPRK, the IAEA has sought to use other sources of information, including trade-related information, as indicators of undeclared safeguards relevant activities. This new strategy took shape in two ways.

Firstly, the voluntary reporting scheme (VRS) endorsed by the IAEA Board of Governors in 1993 provided a scheme for voluntary reporting by States of exports, imports and production of nuclear material and exports and imports of specified equipment and non-nuclear material. Later on, the additional protocol (AP) [2] included requirements for new information to cover imports and exports of source material holdings, and information on exports of specialized equipment and material specified in Annex II of the AP, as well as a description of a number of activities specified in Annex I of the AP.

Secondly, the IAEA started to use information from open and other sources, including trade information, to reveal indicators of undeclared or incorrectly declared activities. The establishment of the IAEA Trade and Technology Analysis Unit (TTA) in the Department of Safeguards in 2004 has been instrumental in implementing this strategy.

Since 2005, IAEA General Conference resolutions have repeatedly called upon all States to support the Secretariat's efforts to verify and analyze information on nuclear supply and procurement provided by Member States thus providing a clear mandate [3]. Further, the IAEA Medium Term Strategy 2006-2011 [4] clearly stressed that identifying new sources of trade-related information relevant to safeguards is a priority for the IAEA.

Collection and analysis of trade related information is potentially useful for:

- The State evaluation process and for drawing broader safeguards conclusions[1];
- Verifying import and export declarations made by States under APs [2];
- Identifying indicators of activities to be declared under APs [2].

To serve these purposes, the IAEA is using State declared information (e.g., export declarations made by States under APs or CSAs), open source information, as well as information provided by States on a voluntary basis. Each source of information presents strengths but also limitations.

To fill knowledge gaps and cross-check information derived from different sources, the IAEA launched a procurement outreach programme [5][6], started developing new information analysis tools and methodologies [7] and improving understanding of the market of sensitive nuclear technologies. Exploring the use of world trade data for safeguards purposes is a continuation of these endeavours.

In this paper we investigate the value of using trade data services to support safeguards verification activities. Section 2 of the paper reviews services available on world trade data with an emphasis on geographical and temporal coverage, type of data provided and accessibility. Since trade data in data services is reported according to the HS nomenclature of goods [8], a prerequisite to use this type of data is to relate items of safeguards interest with HS categories. This is addressed in Section 3. Section 4 presents a general methodology of how statistical trade data could be used for nuclear verification purposes. Finally, Section 5 concludes the paper with a comparison of the features of the information acquired from trade data services with the features of other information sources used by the IAEA and a general discussion on the overall value and limitations of using trade data services in support to the safeguards verifications activities.

2. Data services on the world trade

The data considered are open source and regulatory in nature: they stem from declarations made by importers/exporters to national customs authorities. The data are collected at national level and, by decision of individual States, released under specific formats, either for free or through pay services.

Services on trade data fall in two main categories: services on transactions and services on statistical data.

2.1 Services on transactional data

Services on transactions are collections of data largely equivalent to original declarations made by importers/exporters to customs authorities.

[1] For each State with a safeguards agreement in force, a *conclusion* is drawn on an annual basis, relating to the non-diversion of nuclear material which has been placed under safeguards. For each State with a CSA and AP in force, a *broader conclusion* can be drawn that all of the nuclear material in the State had been placed under safeguards and remained in peaceful nuclear activities or was otherwise adequately accounted for.

Declared data fields subject to disclosure may include:

- A code classifying the commodity traded (e.g. according to the HS [8] product nomenclature, see Section 3);
- Free text description of the commodity;
- Quantity, expressed in weight or number of items;
- Value;
- Date of shipment;
- Country/port of import/export;
- Party names (importer/exporter).

Some services provide additional information on the shipment's routing, such as the ID of the container that transported the goods, the bill of lading, and the vessel name.

The scope of transactional services can be national or multi-national. For countries where data related to customs declarations are released, multiple services exist offering this data in various combinations. Figure 1 shows a map of countries for which collections of transactional data have been found to exist as a result of a survey conducted by JRC in 2007. For each country, an indication of the number of years of data available is shown. The earliest transactional data available starts in 1995 (for the United States, on imports only), but many States have started to release transactional data only in the last five years. It must be noted that the number of States publishing customs data is not necessarily increasing in a steady way: a State may change its policy for the dissemination of this data or considerably reduce its scope by suppressing key fields.

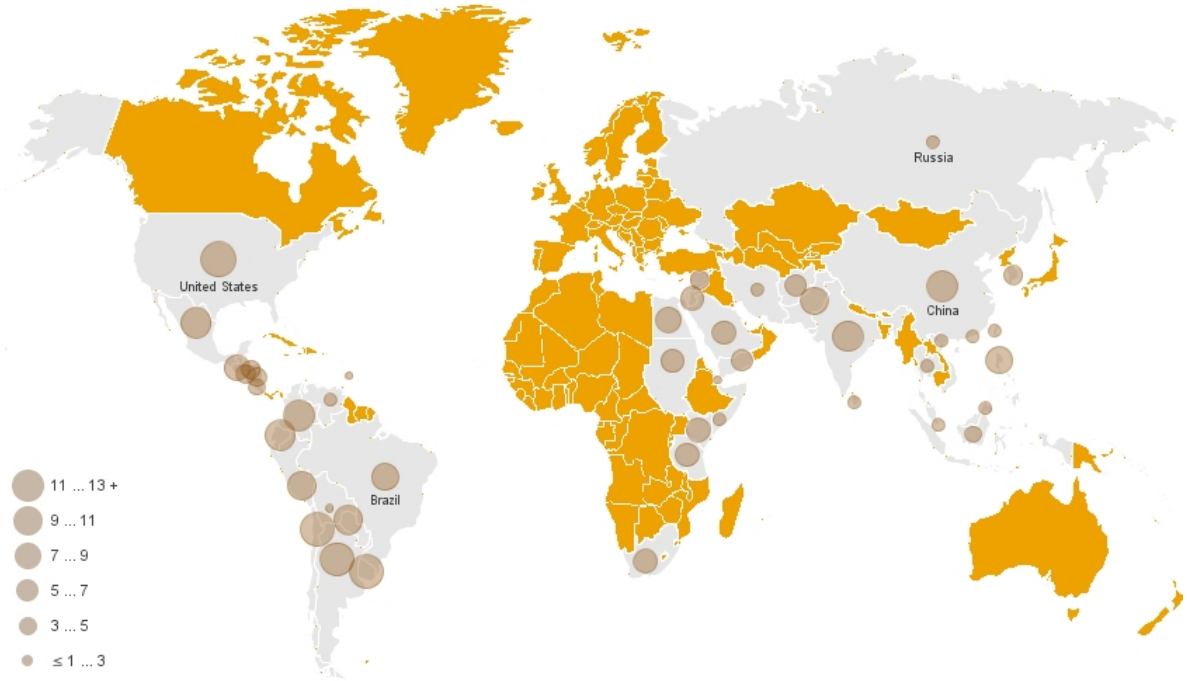

Figure 1 – Services on trade transactions geo-located. For each country where transactional data has been found (countries coloured grey), the number of years of data available in 2007 is shown.

Services on transactional data are offered mostly by private companies against a subscription fee. This can be conspicuous, especially for services with a multi-national scope and offering a single interface point that allows querying many data sources at once. In general, cost is a function of the data sources included in the service: the number of data fields, the number of product categories and country sources. Some services offer predefined combinations of data sets (with a fixed cost service), others allow for customized combinations of data sources.

Transactional data services can be delivered online or off-line. Online services are the most common, and rarely offer archive data (e.g. data before 2004). For archive data, CD-ROM services are offered instead.

2.2 Services on statistical data

Statistical data on trade is derived by aggregating transactions data by country, trade flow (import or export), period of time (months, years) and product categories as specified by the adopted nomenclature, the most common being the HS [8] .

Typically, a data record includes:

- Reporting country, this can be either the exporting or the importing country;
- Partner country in trade;
- Trade flow (import or export);
- Category of commodities;
- Time period (months or years);
- Cumulative value of the trade for the above fields;
- Cumulative quantity of trade for the above fields.

The scope of statistical data services is in most cases multi-national. As an example, COMTRADE [9], provided by the United Nations, offers the largest geographical coverage, including 150 reporting countries with annual series of data, and archives dating back to 1995 or earlier. COMEXT [10], provided by EUROSTAT, the European Union Statistical Office, is a second example: focused on EU reporting countries, it provides monthly records of trade data since 1995. Being based on monthly records, COMEXT provides shorter timeliness than COMTRADE. On the other hand, due to the higher number of reporting countries, COMTRADE gives a more global perspective on the world trade. Because COMTRADE includes bi-lateral declarations provided by countries which are partners in trade (imports versus exports), it is possible to estimate missing data by mirroring the statistics between partner countries, or analyse with priority records provided by countries whose reports appear to be more reliable.

Statistical trade data is mainly offered by international organizations (e.g. United Nations, European Commission), governmental organizations and national statistical offices, often for free or for limited fees. The data collection supporting these services takes place at national level, together with the aggregation of the data. After aggregation, the national data is released in broader data services contributing to building a global perspective on trade (e.g., as in COMEXT and COMTRADE). Statistical data are shared by respecting data confidentiality requirements whose definition are country-specific. In the EU and in a number of other countries, the data used for the production of statistics are considered confidential when it allows statistical units to be identified (e.g., the value of a single shipment) either directly or indirectly [11]. Because the precise operational criteria used to decide which statistical data are to be considered confidential are fixed by national legislation, an important part of any data service is to provide the users with ‘meta-data information’ which documents the procedure for the data collection and data treatment before this is released.

3. Mapping items of interest to safeguards verification with HS codes

Most data available from trade data services are retrieved by product categories, the most commonly used being those of the HS [8]. Designed by the World Customs Organization, HS has become the reference taxonomy for commodities adopted (besides customs) by trade associations and statistical offices in the majority of countries.

HS is based on about 5,000 commodity groups organized within 22 Sections in a hierarchy made up of:

- Chapters;
- Headings;
- Subheadings.

Each level in the hierarchy is identified by a HS code and an explanatory note. Codes are 2-digit for Chapters, 4-digit for Headings and 6-digit for Subheadings.

For example, the following sequence leads to a six-digit HS code for milling machines[2]:

84 → 8459 → 8459.61

84	*NUCLEAR REACTORS, BOILERS, MACHINERY AND MECHANICAL APPLIANCES; PARTS THEREOF*;
8459.	*Machine-tools, incl. way-type unit head machines, for drilling, boring, milling, threading or tapping (excl. lathes and turning centres of heading 8458, gear cutting machines of heading 8461 and hand-operated machines)*;
8459.61	*Milling machines for metals, numerically controlled (excl. way-type unit head machines, boring-milling machines, knee-type milling machines and gear cutting machines)*.

Since most trade data available through data services (whether statistical or transactional) are reported by HS, a precondition to access relevant data is to 'map' items of interest with HS codes.

Generally, two approaches are possible.

The first is to browse the HS guided by its hierarchical structure or through a textual search on keywords to identify HS codes that describe items of interest, as shown in the example above.

The second approach is to consult existing correspondence tables compiled by experts that associate HS codes to items subject, for example, to export controls regulations. One such a table is the so-called Correlation Table, [12][13] mapping to HS items listed in the European Union (EU) Council Regulation setting up a Community regime for the control of exports of dual-use items and technology [14][15]. The Correlation Table is part of the Integrated tariff of the European Communities or TARIC [16]. TARIC incorporates the Community legislation on trade concerning tariff suspensions, quotas, import/export prohibitions, surveillance, restrictions, etc. Within TARIC, the Correlation Table serves the practical purpose of informing exporters as well as customs officers in EU Member States about restrictions that apply to the trade of goods listed in the EU Council Regulation on dual-use, which includes, among others, items subject to nuclear export controls[3].

Following the previous example, Table 1 shows some of the HS codes associated to 'milling machines' by the Correlation Table. Querying TARIC for existing trade restrictions in the EU on all these HS codes retrieves a reference to the particular controlled item listed in Annex I to the EU Council Regulation on dual-use items and technology, in this case item 2B001b, whose definition is shown in Figure 2.

[2] Certain high-precision milling machines can be used for the manufacture of centrifuges.

[3] Annex I of EU Council Regulation on dual-use provides a single list of dual-use items and technology drawn from export controls lists to which EU Member States adhere. These are: the Wassenaar Arrangement, the Missile Technology Control Regime, the Australia Group, the Chemical Weapons Convention and the Nuclear Suppliers Group.

Product code	Meaning
8457.10	Machining Centres, For Working Metal
8457.20	Unit Construction Machines (Single Stage) For Working Metal
8457.30	Multi-Station Transfer Machines For Working Metal
8459.10	Way-Type Unit Head Machines For Removing Metal
8459.31	Boring-Milling Machines Nes, Numerically Controlled For Removing Metal
8459.51	Milling Machines, Knee-Type Numerically Controlled For Removing Metal
8459.61	Milling Machines not elsewhere specified, Numerically Controlled For Removing Metal

Table 1 – Examples of HS codes corresponding to 'milling machines' according to the Correlation Table.

Machine tools for milling, having any of the following characteristics:

1. Having all of the following:

   a. Positioning accuracy with "all compensations available" equal to or less (better) than 6 µm according to ISO 230/2 (1988) (1) or national equivalents along any linear axis; and

   b. Three linear axes plus one rotary axis which can be coordinated simultaneously for "contouring control";

2. Five or more axes which can be coordinated simultaneously for "contouring control";

3. A positioning accuracy for jig boring machines, with all "compensations available", equal to or less (better) than 4 µm according to ISO 230/2 (1988) (1) or national equivalents along any linear axis; or

4. Fly cutting machines, having all of the following characteristics:

   a. Spindle "run-out" and "camming" less (better) than 0,0004 mm TIR; and

   b. Angular deviation of slide movement (yaw, pitch and roll) less (better) than 2 seconds of arc, TIR over 300 mm of travel.

Figure 2 – Definition of 'machine tools for milling' as in the EU Council Regulation on dual-use items and technology [15].

Comparing the detailed definition in Figure 2 with the descriptions provided by HS codes in Table 1 highlights the degree of approximation introduced by the use of these codes, which do not discriminate between machines usable for nuclear purposes from other machines. Despite that, these HS codes are the ones most likely used by exporters and importers when declaring trade of high-precision milling machines. Under this hypothesis, their trade shall appear under these categories in

databases derived from customs declarations, together with the trade of milling machines intended for other uses. One can expect favourable cases where actual trade on machines tools with nuclear end-use reported by these 'generic' HS codes can be recognized by additional criteria, such as the trade unit value[4]. Note that the problem of 'disambiguating' trade data reported by HS codes is also faced by customs authorities whose first, indicative 'risk assessment' is based on HS codes, together with value and quantity indicators.

From a technical safeguards perspective, it is recognized that the correspondence between items of interest to safeguards verification and HS codes is of uneven quality and weak for several items. As such, correspondence tables can only provide an indication on which HS categories trade of interest might have been declared to customs authorities. Nevertheless, there can be circumstances under which the use of HS codes to retrieve open source data about trade can be relevant for IAEA safeguards purposes.

4. A possible use of statistical trade data services for safeguards verifications

In the framework of the EC Support Programme to the IAEA, the JRC is conducting feasibility tests to assist the IAEA in the evaluation of possible uses of trade data services in support of safeguards verifications.

A sample test starts by a piece of information to be verified by the IAEA, for example, the export of a nuclear material from a country in a given timeframe. The goal of test is to retrieve relevant data from sources on trade (presented in Section 2) for the IAEA to verify the correctness and completeness of the initial information.

Before consulting sources on trade, items of interest to the test case need to be clearly identified. This step is supported by trade and technology experts' advice and a set of reference documents where materials and technologies of interest are listed and described (e.g. [2]). These items are then 'translated' into HS codes by correspondence tables and direct searches on the HS as explained in Section 3.

Having selected relevant HS codes, a plan of queries is designed. Parameters that come into play in the planning are the geographical coverage of the test, the time frame addressed, the type of the data required (i.e., transactional or statistical), as well as the cost (if any) of accessing the data services and their usability.

For example, a typical query on a statistical trade data service would require the specification of the following dimensions:

- A reporting country;
- The HS codes related to the items of interest;
- A trade flow (import or export);
- A time period.

As a result of the query, the data service returns the list of partner countries for which trade on those dimensions has been declared: this is specified by quantity and value attributes to the trade.

Results of queries are then to be analysed by criteria specific to the test case. The analysis leads to the identification of a limited set of points of interest in the retrieved trade data. These data may confirm information known to the IAEA (completely or partially) and provide insights about its completeness.

[4] Milling machines for nuclear end-use have characteristics and precision requirements that increase their cost.

5. Discussion and conclusions

Trade data services are to be seen as a complement to other sources of trade-related information used by safeguards. As an illustration, Table 2 summarizes, in a comparative way, some features of (i) trade information as declared by member States to IAEA, (ii) trade information derived from open source, and (iii) from the trade data services presented in Section 2.

Features	Information from declarations to IAEA	Information retrieved from open source	Trade data retrieved from trade data services
Source	State authorities.	Non-proliferation community, news media, etc.	State / customs authorities.
Nature of information	Official.	Non official.[5]	Official.
Geographical coverage	States with safeguards agreements in force.	All States, with an emphasis on States receiving media coverage.	All States inserted in the world economy, including the 150 States members to the World Customs Organization.
Items coverage	Nuclear material, 'Trigger List'.	All commodities.	All commodities.
Data type	Structured, referenced by IAEA legal definitions.	Unstructured, not referenced.	Structured, referenced by HS codes.
Temporal coverage	Starting from entry into force of safeguards agreements.	No limit.	Since the existence of electronic data bases. More recent (5-10 years) for detailed information.
Information update	Initial, yearly or quarterly declarations.	N/A.	Yearly, quarterly, monthly or by shipment
Continuity	Regular.	Irregular.	Regular.
Import / export mirroring	For CSA and VRS: imports and corresponding exports are declared independently. For AP Annex II items: systematic declarations are due for exports only.	Often reflects one-sided information.	Imports and corresponding exports declared independently.
Reliability	*High* to verify correctness, *medium* to verify completeness.	*Medium* to *low* depending on the source.	*Medium* to *low*, depending on item types, value, trade flow, declaring State.
Accuracy	*High*	*Medium* to *low* depending on the source.	*Medium* to *low*, depending on item types.

Table 2 – Comparative table on usual features (first column) of trade sources stemming from States declarations (second column), open source information (third column) and trade data services (fourth column).

[5] Some open source information can be considered as official, such as public court indictments.

Trade data services present several notable features. *Firstly*, differently from other open source information about trade published by news and media, the data provided by these services are 'official', because they stem from declarations made by exporters and importers to customs. As such, the IAEA can, if needed, discuss these data records with competent States' authorities. *Secondly*, the data cover all commodities in trade; they are not limited to items to be declared to the IAEA under safeguards agreements. This feature may allow for the testing of hypotheses on possible undeclared activities whose indicators may be based on relevant commodities other than those declared by States to the IAEA. *Thirdly*, the data provided are of a quantitative nature (value and quantity of trade), making them more suitable to analysis than open source trade literature. *Fourthly*, the coverage of statistical data is worldwide, a fact that opens the possibility of mirroring declarations between partner countries in trade to complement missing data, or to compensate for reporting countries whose trade declarations appear to be less reliable. Moreover, trade data services may cover countries which do not have additional protocols in force and, as such, do not provide information on nuclear-related imports and exports to the IAEA. *Fifthly*, for statistical data services the collection of records is generally complete over time: time series over more than ten years can be retrieved from data services and analysed.

A limiting factor in the use of trade data services appears to be in the granularity of the data. For several items of interest to Safeguards verifications, HS codes may simply prove to be too generic to retrieve relevant trade data in a reliable and accurate way. On the other hand, there can be other means to 'disambiguate' the data retrieved by these codes (e.g., by indicators on the value of trade, by time series analysis, etc.). For other items, HS descriptions seem to be sufficiently accurate to provide medium accurate indicators about trade.

More specifically, initial feasibility studies on the use of trade data services suggest its Safeguards relevance along the following lines:

- *Supporting the State evaluation process and the drawing of broader Safeguards conclusions* - To reach the conclusion that there are no undeclared nuclear material or activities in a State, the IAEA collects and evaluates information that can be directly or indirectly relevant to the implementation of Safeguards. Trade information could support the evaluation of certain aspects (i) of industrial capabilities by assessing exports, (ii) of international cooperation by identifying trade flows between partner States, (iii) of mining related activities by the exports of raw materials and semi-finished products, (iv) as well as on the general nuclear fuel cycle by imports and exports of nuclear materials and equipment.

- *Verify import and export declarations made by States under APs* - Trade data bases may prove useful to identify trade flows of raw material subject to Safeguards. HS categories appear to be less specific than Safeguards categories, but precise enough to be determined as Safeguards-relevant. AP Annex II items seem more difficult to identify through HS categories, with the notable exceptions of nuclear reactors and parts.

- *Identifying indicators of activities to be safeguarded or to be declared under APs* – In this context it is foreseen that trade data bases could be used to verify hypotheses. Commodities to serve as indicators and methodologies would then need to be identified on a case by case basis and in a hypothesis-specific way.

As a general conclusion, trade data services are not expected to provide on their own evidence of Safeguards-relevant trade or activities, but indications complementing other sources of information received by the IAEA. They can become fully relevant when part of a general analysis strategy, in line with the IAEA Safeguards mandate.

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Export Control Regimes: Emerging International Standards ?

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Introduction

Since the discovery of nuclear power more than a half century ago, the development of different peaceful applications thereof and, in particular, of power generation has always been constrained by the risk of diversion to the military applications. Therefore, numerous export control regimes, notably CoCom, NPT, Zangger Committee, Nuclear Suppliers Group, have tried to establish a set of principles and parameters for the purpose of resolution of this dilemma. Generally speaking the nuclear field is constrained, on one hand, by the politically binding instruments, also known as soft law, adopted by a group of States being usually the main suppliers and, on the other hand, by a small number of legally binding instruments endorsed by international or regional organisations.

Facing the common trend towards deregulation and lifting of international trade barriers for most goods and services, the elaboration and the strengthening of international standards striving for control of transfers of nuclear equipments, materials and technologies constitutes a strange paradox that we intend to analyse in the present contribution.

Precursors of International Export Control Standards: heterogeneous system of successive layers

Since the beginning of the fifties, international trade agreements have always tended to reduce trade restrictions established unilaterally by States aiming at control of flow of goods coming, leaving or passing through their territory. Various negotiations and agreements concluded since the first General Agreement on Tariffs and Trade Round (GATT) adopted in 1948 until the present Doha Agenda have more or less succeeded to substitute a set of international trade rules for national restriction regulations.

Nevertheless, if for common goods and services, such as textiles, air transports, telecommunications, lifting trade barriers appeared necessary to establish an international trading system, it was also acknowledged that other goods and services, notably weapons, dual-use items, diamonds, cultural goods, could not be considered as ordinary goods and for political reasons had to be submitted to or maintained under the national restrictive measures.

Therefore, a general exception and a security exception have been introduced in the GATT Agreement allowing participating States to maintain or adopt national restrictive measures with respect to strategic items[1]. These exceptions are instituted by Article XX (general exceptions) and Article XXI (security exceptions) which could be divided in different groups as regards the items concerned or the political objective pursued by the exception under consideration.

[1] Similarly worded exceptions are contained in articles XIV and XIVbis of the General Agreement on Tariffs and Services (GATS).

As concerns nuclear trade, Article XXI essentially establishes three exceptions which might be eventually implemented by States.

The first exception is clearly dedicated to nuclear area; however it concerns exclusively ***fissionable materials or materials from which they are derived***. This exception might appear amazing as long as it ascertains the nuclear as the only energy source to benefit from GATT security exception which is principally due to historical reasons. Indeed, in 1947 when the agreement was negotiated, only two years had elapsed since the explosion of the first atomic bomb and fissile materials were considered as those having essentially military applications. Moreover, it was thought that the quantity of fissile materials available in the world might be rather limited and mainly concentrated in a few countries such as Canada, Congo and Australia. Therefore, even if the use thereof as a source to produce electricity was known, fissile materials chiefly appeared to negotiators as strategic rather than peaceful goods and were assimilated to a potential implement of war thereby benefiting from GATT derogations.

The second exception concerns the implementation of Contracting Parties commitments ***under the United Nations Charter for the maintenance of peace and security***. Substantially this provision is an implementation of Article 49 of UN Carter which allows the Security Council to decide which measures, other than the use of armed force, are to be employed for the purpose of giving effect to Council decisions. Those measures may, inter alia, include a complete or partial interruption of economic relations, notably a ban on exports of equipment for internal repression, a ban on provision of certain services, restrictions on admission, freezing of funds and economic resources of certain persons, an import ban on diamonds, etc. Besides the imposition of prohibition on export of specific goods and services enables to constrain the States to withdraw doubtful decisions, to adopt acts or to align policies which cause or might cause a breach of peace and security. Such export limitation is prima facie in perfect contradiction with GATT principles that is why it was necessary to supplement the Agreement with dedicated provisions thereby allowing the adoption of trade restriction by States Parties.

As concerns the national implementation of this exception, it is usually mentioned by Contracting Parties in their licensing notification documents which main purpose is to describe rules and procedures imposing import or export prohibition of goods intended for or acquired from certain countries. For instance, "*Brazil's 1994 notification on import licensing notes that the import licensing system of Brazil applies for good entering from or exported to any country except for those covered by UN embargoes. The import licensing notification of Cyprus similarly notes that imports from certain countries are prohibited in accordance with United Nations resolutions*"[2].

Finally the third and the cardinal exception involves the ***traffic in arms, ammunition and implements of war and or such traffic in others goods and materials as is carried on directly or indirectly for the purpose of supplying a military establishment.*** The exact meaning of the terms *arms* as well as *implemented of war* raised several controversies among Contracting States Parties. As regards the term *implement of war*, it was specifically used by the United States at the beginning of the Cold War in order to impose an embargo on export of strategic goods to Czechoslovakia. The list of items concerned included more than 200 entries which appeared for Czechoslovak authorities so extensive that it was hardly possible to define which categories of items it exactly covered[3]. Nevertheless, "*[i]n the discussion the Chairman indicated that Article XXI 'embodied exception to the general rule contained in Article I'. It was stated, inter alia, that 'every country must be the judge in the last resort on questions relating to its own security' (CP.3/SR.22, page 7). The claim that the United*

[2] Article XXI: Security Exceptions, *GATT Analytical Index : Guide to Law and Practice*, p. 605. This document is available at the following website: http://www.wto.org/english/res_e/booksp_e/gatt_ai_e/gatt_ai_e.htm.
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States had failed to carry out its obligations under the General Agreement through its administration of the issuance of export licences, was rejected by roll-call vote (17 to 1, with 3 abstentions) (CP.3/SR.22, page 9)"[4].

The extensive approach of the list of items that could fall under this exception seems to have been the one praised by the negotiators as long as during the discussion a question was brought up, notably *"whether the phrase 'for the purpose of supplying a military establishment' would permit restrictions on the export of iron ore when it was believed that the ore would be used by ordinary smelting works and ultimately for military purposes by another country. It was stated in response that 'if a Member exporting commodities is satisfied that the purpose of the transaction has to supply a military establishment, immediately or ultimately, this language would cover it'"*[5]. Indeed, the majority of States has endorsed a set of restrictive rules regarding the export of strategic goods that comprised a lengthy list of items to control. Such lists contained not only items which indisputably constitute an implement of war, such as machine guns or fighting aircrafts, but also items which are not considered as such weapons but could contribute to the development or manufacture of implements of war as, for example certain lasers, digital computers or telecommunication systems.

It shall be emphasised that these lists usually represent national implementation of strategic goods lists adopted within the bounds of informal agreements gathering certain States which attempt to coordinate the export control policy against third States staying aside from the agreement in question. Initially, the control of strategic goods was suggested by the United States for the purpose of countering the risk of US technology transfer to a Warsaw Pact Members or to the other sensitive countries, such as China. This system called "Coordinating Committee for Multilateral Export Controls" (CoCom) was created in 1950; the main principle thereof was to ban the export of sensitive items, substantially military related ones, to the Soviet Union and its allies. Similar mechanisms dedicated to the fight against the proliferation of weapons of mass destruction (WMD) have been adopted ever since, furthermore the influence thereof upon on international trade have never decreased.

Thus, through the creation of CoCom and the adoption of Atoms for Peace Plan (1953), States have marked their willingness to coordinate national export control policies. The harmonisation was undertaken by the way of elaboration of either legally and politically binding instruments. The result of this harmonisation implies a heterogeneous system established by successive layers of international instruments and leads to a continuous mutation of the export control regimes which become less technical and more political.

Emerging International Export Control Standards

The general implementation field which could be considered as the first step towards the International Export Control Standard might be enacted by the adoption of a common list of nuclear items that States commit themselves to control in respect of their NPT engagements. Indeed, Article III.2 thereof constraints States Parties to "*not provide: (a) source or special fissionable material, or (b) equipment or material especially designed or prepared for the processing, use or production of special fissionable material, to any non-nuclear-weapon State for peaceful purposes unless the source or special fissionable material shall be subject to safeguards*". Therefore, several nuclear export control lists were adopted by essentially two informal international regimes which are the Zangger Committee[6] and the Nuclear Suppliers Group (NSG)[7]. Consequently, if an item figures on one of the abovementioned lists, the supplier State is compelled to require the recipient to implement a comprehensive safeguards

[4] Article XXI note by the Secretariat, 18 August 1987 (MTN.GNG/NG7/W/16).
[5] Article XXI: Security Exceptions, *op.cit.*, p. 603.
[6] Zangger Committee established a list of nuclear items in regard to the NPT definition (INFCIRC 209.Rev.2.mod.1) including Memorandum A (nuclear materials) and Memorandum B (material and equipment).
[7] NSG established a so-called trigger list, which contains 7 categories of items and related technologies to be controlled by State parties.

agreement as defined and implemented by the IAEA. Adopted in 1976 this mechanism triggers the safeguards requirement and seems to be reckoned by the majority of States as an International Standard on Export Control. However this approach has several inconveniences, notably the legal value of such lists remains uncertain; the inadequacy of technical parameters enables the import of items having technical parameters just inferior to the one of the controlled items; and finally the delays necessary for updating the list are excessively long.

Nevertheless, the export control policy implying the list of controlled items initiated by the Zangger Committee and the NSG appears to be commonly accepted. Besides in 1992 the control list has been extended by the NSG to dual-use items defined as equipment, material and technology which have ***both*** nuclear and non-nuclear applications and could make a significant contribution to an unsafeguarded nuclear fuel cycle or to a nuclear explosive.

Therefore, even if the legitimacy thereof has been contested by non-participating States, it could be wondered at what measure NSG Guidelines might be considered as a set of International Export Standard? Firstly, a positive response inevitably comes to mind. Indeed, the control of dual-use items and technologies seems to be in line with the obligations endorsed under UN Security Council Resolution 1540. In addition, the fact that both nuclear and dual-use NSG lists were included in Guidelines published by the IAEA enlarges considerably a scope of application thereof. Even if it is true that there are still a few potential and existing nuclear supplier States that remain aside the NSG agreement, such as India, Pakistan, Israel and North Korea, it seems than non-NSG member States gradually align their formal or informal national policies to NSG Guidelines provisions.

Even though International Standards do exist through NSG Guidelines, a reflection shall be deepened by the analysis of implementation thereof by States Parties. Therefore, four cases will be performed in order to show that the implementation of international commitments usually proves to be unequal.

Firstly, ***an interpretation of NSG Safety clause*** which makes each supply of nuclear items conditional upon bringing into force by the end-user State of an agreement with IAEA imposing the application of Comprehensive Safeguards Agreement. In should be noted that two exceptions were introduced, in particular a “grandfather clause” permitting the completion of commitments of the supplier State linked to contracts signed before its NSG membership. In addition, a “safety clause” was foreseen in order to allow transfers to a non-nuclear-weapon States when they are deemed essential for the safe operation of existing facilities and only if safeguards are applied to those facilities. Russia was the only State to apply safety clause for the purpose of supply of fuel assemblies for the Indian Tarrapur nuclear plant. Those transfers were strongly criticised by the majority of NSG States Parties which tabled several proposals aiming at reinforcement of the provision. Nevertheless, this incident has shown that the interpretation of the safety clause can easily exceed the initial meaning thereof.

Secondly, ***the implementation of NSG trigger and dual-use lists*** by the European Union via Council Regulation (EC) No 428/2009 establishing a unique list of dual-use items requiring export or transfer authorisations[8]. Even if the abovementioned list is considered as comprehensive and compulsory, some Member States consider that it shall be submitted to national appreciation or interpretation concerning whether an item should be or not included on the list. Thus, some items and technologies are submitted to export authorisation in some Member States and not in the others.

Thirdly, ***the implementation of catch-all clause*** by the European Union through Article 4 of Regulation No 428/2009 which institutes three different catch-all clauses increasing the responsibility of exporters in the fight against nuclear weapons proliferation. Indeed, two compulsory catch-all clauses impose an export authorisation for not listed items if, on one hand, an authority deems that

[8] Council Regulation (EC) No 428/2009 of 5 May 2009 setting up a Community regime for the control of exports, transfer, brokering and transit of dual-use items (recast) (OJ L 134, 29.5.2009, p. 1).

there is a risk of misuse thereof for WMD program; or on the other hand, the exporter is aware that item will contribute to such program. In addition, an optional so-called "suspicion clause" was introduced granting a possibility for State to require an export authorisation if the exporter has grounds for suspecting that the item will contribute to WMD program[9]. Thus, the existence of optional catch-all clause as well as the vagueness of vocabulary used by Article 4 results in heterogeneous implementation of catch-all technique by EU Member States.

Fourthly, ***the understanding of concept of control of technology*** does not appear to be uniform among 27 Member States. Generally speaking the definition of technology used by the Regulation No 429/2009 is identical to that employed by the NSG Guidelines even as regards the exception provision[10]. However, the understanding of this provision varies in accordance with EU Member States, for some of them any export of technology will be submitted to authorisation taking into account that industry never conducts a basic research rather aiming at developing of marketable product. All Member States do not share such restrictive vision, that is why the same technology might be subject to authorisation in one State, at exempted therefrom in the other.

Conclusion

The performed analysis of worldwide trends in nuclear trade acknowledges that the Potential International Export Control Standards has been established since early ninetieth. However, as it was ascertained, even if the international rules do prevail, the main issue remains the necessity of harmonisation of implementation thereof by the States Parties. One of the solution might be the establishment of an efficient legally binding *no undercut mechanism*.

Nevertheless, the success of elaboration of common International Standards as well as the issue of international nuclear safety remains largely dependent upon the political will of States.

[9] This catch-all clause was adopted by following Member States: Austria, Cyprus, Czech Republic, Denmark, Estonia, Finland, Greece, Hungary, Luxembourg, Malta, Netherlands, Poland, Slovakia, Spain and United Kingdom.

[10] "*Controls on 'technology' transfer do not apply to information' in the 'public domain' or to 'basic scientific research'"*. See NSG Guideline 254/Rev.9/Part.1 and Nuclear Technology Note, Annex I of Regulation No 1334/2000.

SESSION 13

MEASUREMENT TECHNIQUES FOR SPENT FUEL

Determining Plutonium in Spent Fuel with Nondestructive Assay Techniques

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Abstract:

There are a variety of motivations for quantifying plutonium in used (spent) fuel assemblies by means of nondestructive assay including the following: shipper/receiver difference, input accountability at reprocessing facilities and burnup credit at repositories or fuel storage facilities. Twelve NDA techniques were identified for providing fuel assembly composition information.[1] Unfortunately, none of these techniques, in isolation, is capable of determining the Pu mass in an assembly. However, it is expected that the Pu mass can be quantified by combining a few of the techniques. Determining which techniques to combine and estimating the expected performance of such a system is the purpose of a research effort recently begun. The research presented here is a complimentarily experimental effort. This paper will focus on experimental results of one of the twelve non-destructive assay techniques - passive neutron albedo reactivity. The passive neutron albedo reactivity techniques works by changing the multiplication that the pin experiences between two separate measurements. Since a single spent fuel pin has very little multiplication, this is a challenging measurement situation for the technique. Singles and Doubles neutron count rate were measured at Oak Ridge National Laboratory for three different burnup pins to test the capability of the passive neutron albedo reactivity technique.

Keywords: spent (used) fuel; nondestructive assay

1. Introduction

Although the majority of plutonium (Pu) in the world is stored in commercial spent (used) fuel assemblies, a measurement system for directly quantifying the Pu mass contained in these assemblies does not exist. The nondestructive assay systems in use today (Cerenkov Viewing Device,[2] Fork Detector[3] and Safeguards Mox Python Detector[4]) essentially measure indirect signatures from spent fuel such as gamma emission from fission fragments, or photons induced by radiation from fission fragment, or total neutron emission that pre-dominantly is emitted from curium. Calculation codes, known as burnup codes, can be used to infer plutonium mass from these measured signatures. In order to use burnup codes to predict the Pu mass in a particular assembly, input from the operator is required. From an international safeguards perspective, this input is undesirable given the regulatory requirement of independent verification.

Below, nine reasons for improving on the status quo are listed. These reasons are the motivation for designing a nondestructive assay (NDA) system that can quantify the Pu mass in spent fuel assemblies: (1) Provide regulators with the capability to independently verify the mass of plutonium at any site that has spent fuel. (2) Enable regulators and facilities to accurately quantify the Pu mass leaving one facility and arriving at another facility ("shipper/receiver difference"). (3) Provide confidence to the public that the shipment of spent fuel around the world is being undertaken in a rigorous manner; assure that material is not diverted during shipment. (4) Provide regulators with a tool for recovering continuity of knowledge at any site storing spent fuel. (5) Provide reactor operators

with a tool enabling optimal reloading of reactor cores. (6) Provide regulators of once-through fuel cycle repositories the capability to optimally pack fuel both for transport, in a pool and into the repository ("burnup credit"). (7) Enable determination of the input accountability mass of an electro-chemical (pyro-chemical) processing facility. (8) Provide facility operators with a means for quantifying the Pu mass in spent fuel that is no longer considered "self-protecting." This is particularly relevant given that some regulatory agencies are considering changes to the level at which radioactive material is considered to be self-protecting. And (9) promote cost savings by facilitating assembly selection for reprocessing since facility operators need to combine assemblies to obtain optimal compositions in reprocessing solutions. The blending is presently based on reactor history and burnup codes. The inaccuracy of the status quo decreases plant operational efficiency.

For the purpose of measuring the Pu mass in spent fuel assemblies, 12 NDA techniques were identified.[1] The subject of this paper is one of those techniques – the passive neutron albedo reactivity technique (PNAR). The research presented here adds to recent publications on PNAR[5, 6, 7] in that it is the first time a single spent fuel pin was measured with the PNAR technique.

2. Passive neutron albedo reactivity

2.1. Basic concept

The PNAR technique functions by using the intrinsic neutron emission of the fuel (primarily from the spontaneous fission of curium) to self-interrogate the fissile material in the fuel itself.[5] Two separate measurements of the spent fuel are made, and the ratios of the count rates obtained are analyzed; this ratio is called the cadmium ratio. The primary difference between the two measurements is the neutron energy spectrum and fluence in the spent fuel. By varying the material around the spent fuel, a high and a low neutron-energy-measurement condition is produced. The PNAR technique can be used with total neutrons (singles) and/or doubles and/or triples; it is expected that doubles will produce the best result in the high count-rate regime.[6] If the geometry of the measurement situation is unchanged between two measurements, the change in the cadmium ratio between measurements of different pins or assemblies may be calibrated to a change in the fissile content of the pin or assembly.

One approach to producing these two energy conditions involves measuring the spent fuel with a thin layer of cadmium surrounding it and a second measurement without the cadmium around the fuel. The presence of the cadmium effectively eliminates all neutrons below 1 eV from reflecting back into the fuel from the detector walls, altering the in-leakage reactivity contribution. Hence, in the measurement made with no cadmium present, the fuel is interrogated by all the neutrons reflected back to the fuel. In contrast, when cadmium is present, the fuel is interrogated by only those reflected neutrons with energies above 1 eV. Effectively, how far the ratio deviates from ~1 indicates the impact thermal neutrons.

In order to illustrate this concept further, the results of an earlier publication[5] are reproduced in Fig. 1.

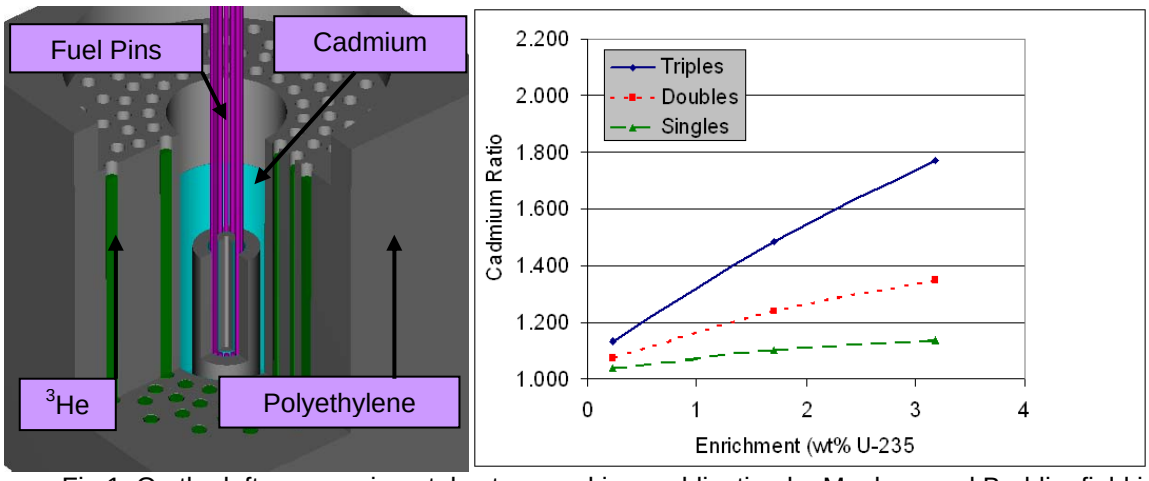

Fig.1. On the left an experimental setup used in a publication by Menlove and Beddingfield is illustrated. On the right, data obtained with the experimental setup is presented.

An interesting ramification of using the neutrons from the spent fuel to interrogate the spent fuel is that the statistics get better for the singles as the inherent neutron source in the spent fuel becomes more intense as compared to many other NDA techniques which need to detect a signal on top of the background.

It is noteworthy that the measurement of one pin is a very sub-optimal geometry for using the PNAR technique. This is due to the fundamental fact that the PNAR concept depends on the change in multiplication between two measurements. There is very little multiplication in one pin, hence the change in multiplication is very small.

2.2. Detector design

The detector utilized in these experiments was a modified Inventory Sample Neutron Coincidence Counter (INVS) detector (16 ^{3}He tubes, 4 atmospheres of pressure, 4 preamplifiers). The cross section and an exterior view of the detector are depicted in Fig. 2. The modification involved the following: (1) removing the polyethylene inside of the ^{3}He tubes, (2) fabricating three concentric inserts, from largest to smallest: lead, poly and cadmium, (3) covering the top and bottom of the detector with lead, (4) covering the entire detector in cadmium. The cadmium liner close to the fuel is removable in order to enable the PNAR technique. The reason the lead and the polyethylene layer are in the order they are, is to maximize the impact of removing the cadmium liner on the neutron energy spectrum in the pin. As modified, the detector is 30% efficient to californium neutrons emitted from the center of the chamber when the cadmium liner is in place. In order to keep the gamma dose to the ^{3}He tubes to an acceptable level, an iron shielding structure was fabricated around the detector; the detector inside the iron shield is depicted in Fig. 3.

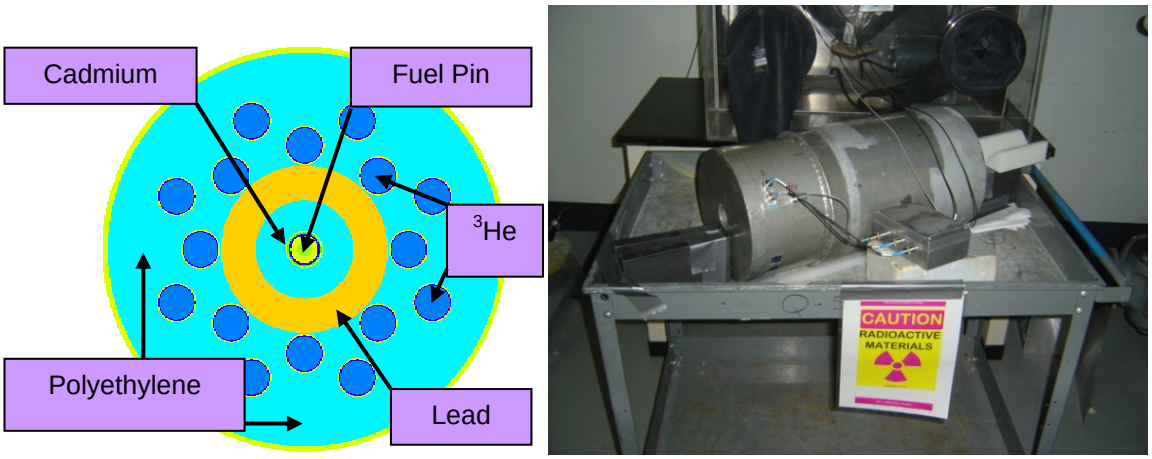

Fig. 2. On the left, the cross section of the detector is illustrated. On the right, an external view of the detector is depicted.

The detector system was designed to allow full length fuel pins to be measured as well as pin segments. In order to measure full length pins, the iron shielding depicted in Fig. 3 was designed to fit over the mechanical fuel drive system used to manipulate full length pins at Oak Ridge National Laboratory (ORNL). The two support structures depicted in Fig. 3 were designed to center the fuel for the measurement of partial length fuel pins. The support was designed so that the Cd liner could be moved in and out of the detector without moving the fuel.

Fig. 3. The iron shielding and fuel pin support structures are depicted.

2.3. Experimental operation

2.3.1 Measurement opportunities

There were three measurement opportunities: May and July of 2008 and January of 2009.

2.3.2 Data acquisition

During the May 2008 measurements, the signal from the detector went to two Advanced Multiplicity Shift Registers (AMSR). The difference between the two AMSRs was only the duration of the gate; one was set to 64 μs while the other was 128 μs. In the later two experimental campaigns, the signal was split as well. However, in order to enable more versatile data analysis, one signal went to an AMSR with a 128 μs gate while the other signal went to a list mode data acquisition system.

The following settings were used for all the data presented in this paper unless stated otherwise: predelay of 4.5 μs, 128 μs gate, 180 ns multiplicity deadtime, deadtime coefficient A of 0.72, deadtime coefficient B of 0.13, doubles gate fraction of 0.65. These values were determined from measurements of ^{252}Cf sources of variable strength and experience with similar detectors.

2.3.3 Description of spent fuel

The fuel measured for this research was at ORNL as part of research being performed for other sponsors. For this reason, the fuel was cut up into segments. The segments used in this publication are listed in Table 1; these segments were between 60 and 75 cm long. The fuel segment for which data is depicted were at least 60 cm from the end of the pin to assure that each pin segment had a relatively uniform axial burnup and to assure that the burnup of the pin segment was close to the average burnup of the assembly.[8, 9] The details of the fissile content calculation are given in section 3.1

Pin Segment	Reactor	Burnup (GWd/tU)	Initial Enrichment	Pin Location	Fissile Content (g/cm)
591D	Surry-2	36.0	3.11%	H9	0.186
616B	Three Mile Island - 1	50.9	4.00%	D5	0.193
651B	North Anna	68.4	4.12%	B16	0.147
652B	North Anna	71.6	4.12%	D5	0.147

Table 1, the pin segment labelling system used by those cutting the pins is listed along with the reactor from which the fuel came, the approximate burnup, and the initial enrichment. The pin location refers to the location of the pin within an assembly.

In order to protect the cladding, each fuel pin segment was placed inside of a steal cylinder. One consequence of this measurement situation is that the fuel had two possible means by which to move

within the steel cylinder. The pellets could move within the cladding. Also the cladding could move within the protective steel cylinder.

2.3.4 Experimental procedure

Given operational constraints, it was not possible to use the mechanical structures designed to center the fuel pin in the detector. This situation means that the fuel had to be moved between each measurement. The intent had been that only the Cd liner was moved between a "*with Cd*" and "*without Cd*" comparison. This movement means that the change in the count rate between two such measurements was due to statistical variation, change in the fissile content as well as a change in positioning. In order to minimize the possibility of the fuel moving inside the cladding or support structure, the fuel was tilted before putting it into the detector to cause any mobile fuel to go to the end of the pin.

Since the fuel measured was in the form of pin segments and since the segments were inside of a protective cylinder and since it was possible for the fuel might move inside the protective cylinder, the location of the fuel was not certain. In order to determine the location of the fuel with the neutron detector, the fuel was pushed through the detector in 5 cm intervals so that the center of the pin could be found.

The first fuel measured was the 70 GWd/tU fuel and had the most intense gamma dose. It was so intense that it was necessary to lower the bias voltage of the ^{3}He tubes to prevent gamma pile up from registering as neutron counts. The detector was 30% efficient for a voltage of 1680 V. For the operating voltage of 1560 V, it was 19% efficient.

3. PNAR Experimental data and analysis

3.1. Data quality check

In order to gain confidence that the detector was operating properly and to confirm that the neutron intensity scaled with burnup as expected, the neutron intensity detected is graphed in Fig. 4 as a function of burnup, or more precisely, as a function of exposure. It is expected that the singles (or total) neutron count rate will scale as the burnup raised to the third or fourth power for burnups above 10 GWd/tU.[10] This scaling is due to the fact that ^{244}Cm is produced at this rate as well as to the fact that ^{244}Cm is responsible for over 95% of the spontaneous fission neutrons for fuel that is cooled for more than 4 years. Slight corrections were made for cooling time and variations in the fuel cross sectional area among the three fuel types. Note that the statistical error bars are much smaller than the width of the data points; a typical uncertainty for the singles count rate was 0.05% as determined by analyzing the scatter in the measured count rate. The scatter was determined by breaking every count time into numerous subintervals. No correction was made for variations in initial enrichment or multiplication since the impact of these factors does not prevent the overall purpose of Fig. 4 which is to gain confidence in the singles neutrons data relative to the documented burnup of each pin.

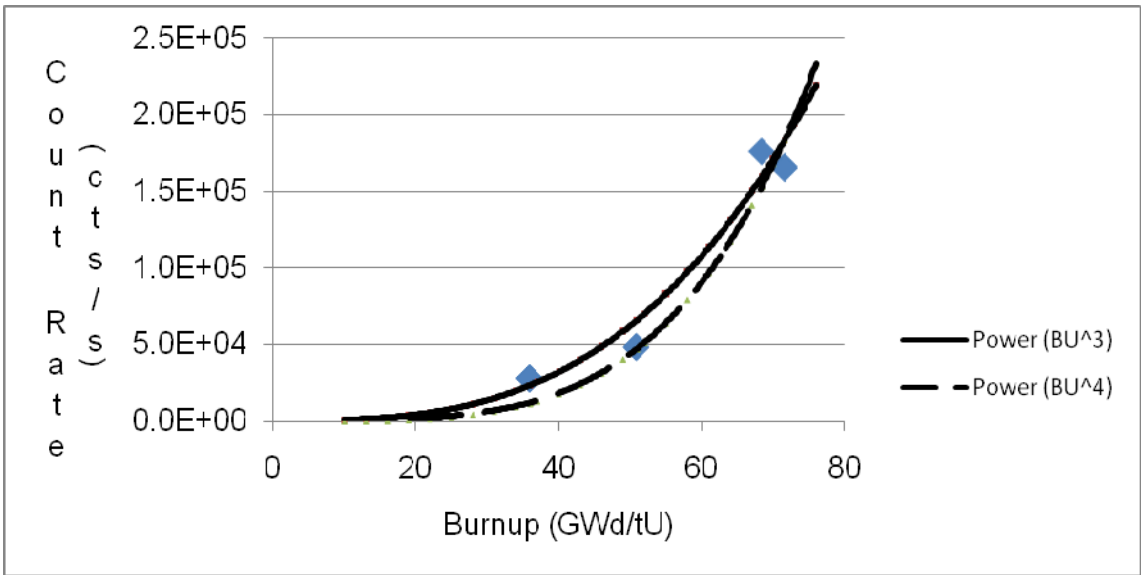

Fig. 4, Singles count rate as a function of burnup. The four data points are for 4 different pins. The two curves indicate the count rate if this rate varied with the burnup to either the third (solid line) or fourth (dashed line) power.

The relationship between the burnup and the singles count rate in Fig. 4 agrees roughly with the expected power scaling. Note that the burnup values used were the values declared by the reactor operators. There was no effort made to correct for variations in the burnup within an assembly. Furthermore, since the fuel came from three different reactors, it is not clear how consistent the techniques for determining burnup were, nor if the timing of assemblies in the reactor were the same. If greater accuracy in burnup is needed, detailed burnup modelling could be done.

The PNAR technique measures fissile content. It does this by changing the energy spectrum of the neutrons reflected back into the fuel. Hence, the PNAR technique is measuring the change in multiplication between two measurements – provided no other factors changed. Another indicator of multiplication is the ratio of doubles to singles count rates for a given measurement. It is interesting to look at this independent measure of multiplication to gain confidence in the data set. In Fig. 5, the ratio of the doubles to singles is given for 14 separate measurements of four separate pins.

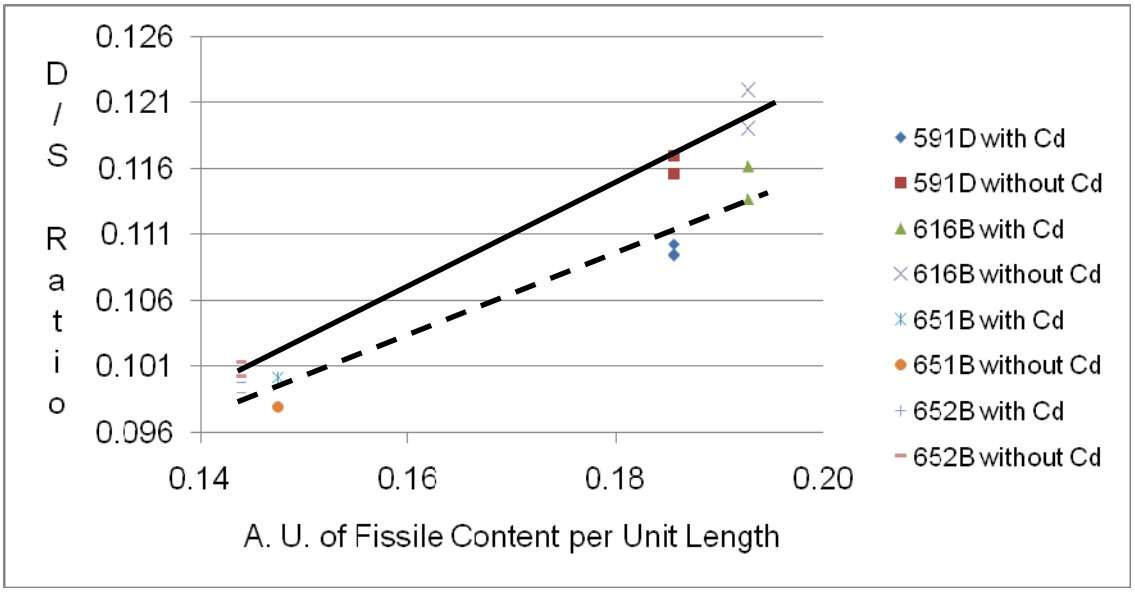

Fig. 5, the ratio of the doubles to singles count rate as a function of fissile content is illustrated for 4 different pins. Each pin was measured with a Cd liner about it (*with Cd*) and with the Cd liner removed (*without Cd*).

The fissile content per unit length illustrated in Fig. 5 was determined by starting with the declared uranium and plutonium fissile mass and isotopic concentrations for ^{235}U, ^{239}Pu and ^{241}Pu. The declared

mass of each fissile isotope was multiplied by the respective thermal cross sections (586 b, 748 b, 1,013 b, respectively) to approximate the fission rate of each isotope (assuming the same neutron flux to each isotope). The thermal cross section is what is of interest with PNAR since the count rate from the non-thermal portion of the spectrum is nearly identical in both the numerator and the denominator of the Cd ratio. In order to weight the fissile content by the number of neutrons produced per isotope, the normalized fission rate was multiplied by the number of neutrons produced per fission (2.41, 2.88, 2.80, respectively) to get the fissile content of a pin. Finally, the total fissile content was divided by the length of each pin to obtain the fissile content per unit length. The lines in Fig. 5 were inserted to roughly separate data points when the Cd liner was not present (solid line) from the cases when the Cd liner was present (dashed line).

In Fig. 5, the lowest fissile content was from the highest burnup cases of ~70 GWd/tU (pins 651B and 652B). In order to discern the data separately from each of these pins, a slight shift in fissile content was made in the graphing. The middle fissile content pin was for the lowest burnup case up of 36 GWd/tU (pin 591D). The highest fissile content was for pin 616B with a burnup of 51 GWd/tU. The one sigma variation in the D/S ratios was dominated by the statistics of the doubles for which the data points as graphed are ~2 sigma (+/- 1 sigma) in width. In the case of 616B (no Cd liner, 10 minute live count time divided into 20 s subintervals), the D/S ratio determined from averaging the three separate measurements of this pin was 0.1210 +/- 0.0003 (Singles = 52,569 cts/s +/- 11.16, Doubles = 6,361 cts/s +/- 16.94).

The primary conclusion from Fig. 5 is that it is possible to detect an increase in multiplication. For 3 of the 4 pins, or 6 of the 7 measurement pairs, the D/S ratio increased with the removal of the Cd liner. The lines that have been inserted were done so to indicate the expected separation between data points when the Cd liner was in place and those when the Cd liner was not present. This is expected since the multiplication should be greater without the Cd liner in place. Pin 651B is the exception. The variation in the S/D for repeat measurements was a little greater than statistics would predict.

Only with pin 651B was the D/S ratio greater with the Cd liner in place than with it absent. It is expected that this result is due to more than just repositioning the pin in the detector. The two data points for this pin are separated by 4 sigma. Based on the other data depicted in Fig. 5, it is expected that the two D/S ratios for pin 651B would have been reversed and separated by a few sigma. Instead they are separated by 4 sigma in the opposite direction. Perhaps the fuel moved within the steal support structure or some other error was made. Note that the D/S ratio is different from the Cd ratio in that both D and S are measured at the same time, hence there is no positioning uncertainty. Yet when comparing two D/S values, there is a positioning uncertainty. It is relevant to note that as a nuclei that fissions moves out of the detector, the probability of producing a doubles count falls off more rapidly than the singles count since the doubles count rate varies as efficiency squared while the singles count rate falls off as the efficiency.

Note that the D/S ratio data in Fig. 5 could be used to quantify the fissile content. If used in this way, one would not need a movable cadmium liner. This approach appears to have greater dispersion and sensitivity than the singles or doubles Cd ratio data for the presently implemented detector. Since the results *without Cd* give a great dispersion, not using a Cd liner would be a preferable means of using this approach.

3.2. Passive neutron albedo reactivity results

In Fig. 6 and Fig. 7 the cadmium ratio as a function of fissile content per unit length is depicted for singles and double count rates, respectively. The data for four pins is depicted. The multiple data points for each pin were determined by taking all possible ratios of *without Cd*/*with Cd*. For example, there are 6 data points for pin 616B since it was measured 3 times *without Cd* and 2 times *with Cd.* Since the goal of presenting the data this way is to indicate the degree of scatter in the data due to positioning, data points were only used if the pins were moved before measurements.

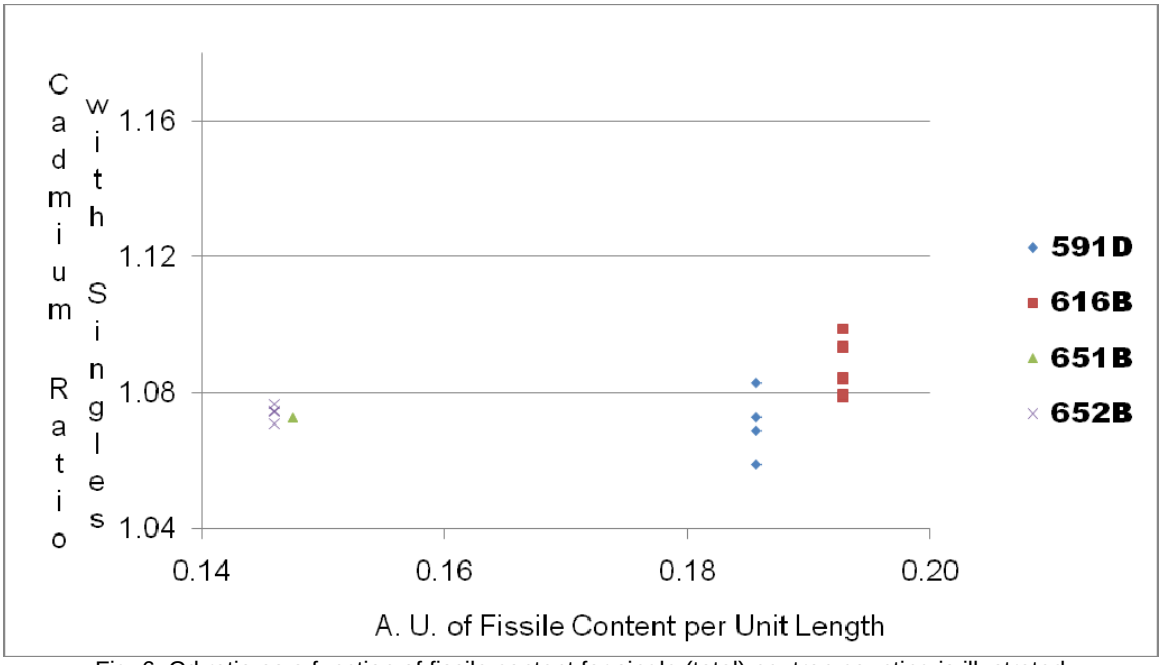

Fig. 6, Cd ratio as a function of fissile content for single (total) neutron counting is illustrated

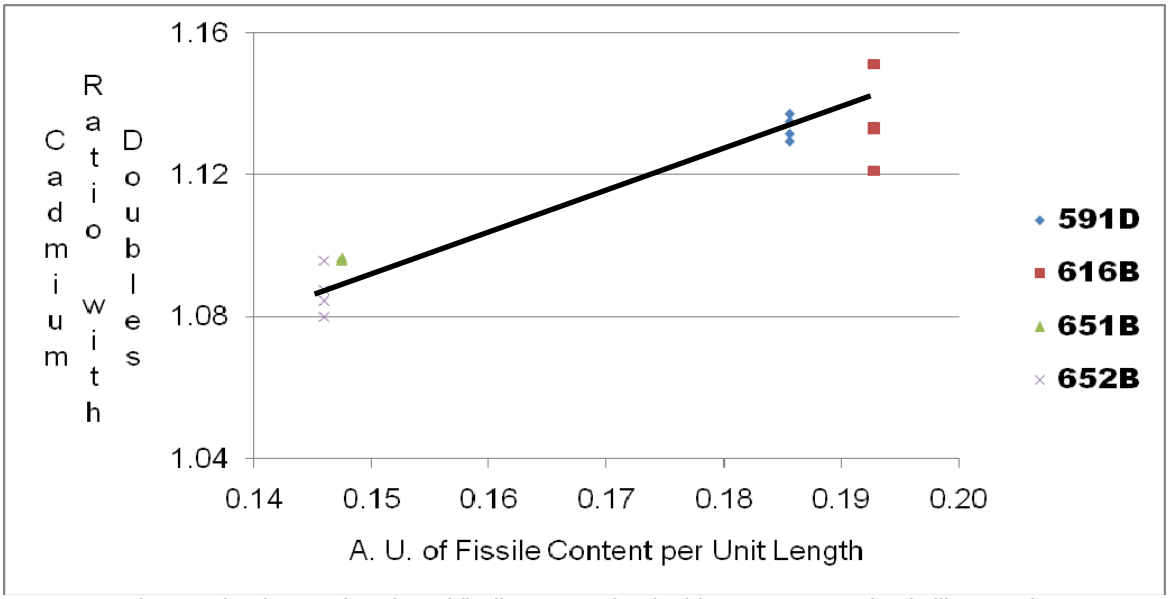

Fig. 7, Cd ratio as a function of fissile content for double neutron counting is illustrated.

Before interpreting Fig. 6 and Fig. 7, it is important to discuss the uncertainties involved. For the singles count rates data of Fig 6, the one sigma uncertainty determined from the scatter in the count rate varied from 0.01% to 0.06%. For the poorest statistical case (pin 591D), the singles ratios was 1.0728 +/- 0.0009. Hence, each data point as graphed in Fig. 6 is ~3 sigma wide. For the doubles rates in Fig. 7, the one sigma uncertainty determined from the scatter in the count rate ranged from 0.2% to 0.6%. For the poorest statistical case (pin 651B), the doubles ratios was 1.097 +/- 0.008. Hence, each data point is ~0.3 sigma wide.

For the singles Cd ratio depicted in Fig. 6, a positive slope with fissile content may exist but the spread in the data for each individual pin is so large that it is rather unlikely that the singles Cd ratio is of much use for quantifying the fissile content in a pin for the experimental setup as it was implemented. It is expected that centering the pin in a more reproducible way and using longer fuel that could not possible move around would improve the singled Cd ratio. It is expected this poorer performance of singles relative to doubles is due to the following two factors: (1) the dispersion in the singles Cd ratio as a function of fissile content is less as depicted in Fig. 1. As a result, the singles Cd ratio will be more sensitive to positioning changes. (2) Variation in the location of the pin in the axial direction of

the detector will produce a greater variation in the singles Cd ratio than the doubles Cd ratio since the doubles count rate falls off more rapidly along the axis as you move out of the detector (i.e. the neutrons coming from outside the active region of the detector will be “filtered” out more effectively by doubles counting than singles). This is because the doubles count rate varies as efficiency squared while the singles count rate fall off as the efficiency.

For the doubles Cd ratio depicted in Fig. 7, a positive slope is clearly present; the current detector as used can detect a change in fissile content for the range of commercial fuel measured. Both improved positioning of the fuel and longer fuel that does not have the possibility of moving will likely improve the results further. However, for the present detector and experimental procedure, the dispersion in the data points for an individual pin is such that a given Cd ratio value corresponds to a wide range of fissile content. To quantify this point, for the three pins that were measured multiple times, the spread in the doubles Cd ratio between the extreme points depicted in Fig. 7 was equivalent to a 2, 3, and 4 sigma variation for pins 591D, 652B, and 616B, respectively. Hence, statistical variation is significant for the doubles Cd ratio even if the uncertainties from experimental procedure were improved.

Counting longer will not improve the results much for the current system. The overnight measurements and several hour measurements were not much better than the ten minute measurements. Drifting in the singles count rate over time was evident in the overnight measurements, because the reduced high voltage was below the plateau for the ^{3}He tubes.

4. Future work

Future work can take one of two paths: (1) research that could be done with the present detector system and (2) research that builds on the lessons learned with the current detector with the focus of designing a new more sensitive detector.

Regarding use of the present system, one could measure full fuel pins. This would minimize the uncertainty due to positioning. Another option is to analyze the list mode data collected to date and possible acquire new list mode data to see if novel analytical techniques can improve the sensitivity of the current detector.[11]

Regarding research that builds on the lessons learned from this work. This path has been developed in the document by Menlove et al.[11] The challenge of applying PNAR in the very non-ideal situation of a single pin resulted in the development of a novel analysis approach and detector design that uses list mode data acquisition to obtain greater fissile content sensitivity. This new detector has a long neutron lifetime near the fuel and a short neutron lifetime near the ^{3}He detectors where the fuel could be either individual pins or full assemblies. The name for this new approach is differential die-away self-interrogation (DDSI). Both DDSI and PNAR are expected to produce significantly better results for a full fuel assembly as compared to single pins because of the increase in multiplication.

5. Summary

A passive neutron detector was modified to enable spent fuel pins to be measured using the passive neutron albedo reactivity technique. This technique uses a movable cadmium liner to change the multiplication in the fuel between two measurements so that the fissile content of the fuel can be quantified by using the inherent neutron emission of the fuel itself to interrogate the fuel. The measurements were made at ORNL with fuel that ranged in burnup from 36 to 70 GWd/tU. From the ratio of the doubles to single count rates, it was clear that a change in multiplication is detected. Both the ratio of the doubles to singles count rates and the doubles Cd ratio allow the fissile content to be quantified experimentally with the ratio of the doubles to singles count rates providing the best results for the detector as designed and implemented. The singles Cd ratio did not provide a good indicator of fissile content. This is thought to be due to the greater sensitivity of this ratio to systematic uncertainties in the experimental procedure. In the context of future work, measuring full length pins would allow some of the experimental uncertainties (moving the fuel between *with Cd* and *without Cd* measurements, uncertainty in the location of fuel, centering of fuel) to be minimized. A novel detector design and analysis approach called differential die-away self-interrogation was conceived in the

course of this work that is expected to improve the sensitivity of a passive neutron detector for quantifying the fissile content of either spent fuel pins or assemblies.

6. Acknowledgements

The authors would like to thank the Safeguards Campaign of the U.S. Office of Nuclear Energy and the U.S. Office of Dismantlement and Transparency (NA-241) for funding this research.

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The Use of Self-Induced XRF to Quantify the Pu Content in PWR Spent Nuclear Fuel

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Abstract:

The development of techniques for the accurate quantification of the plutonium content in spent nuclear fuel would provide significant advances for shipper/receiver differences and for input accountability at reprocessing facilities. Several techniques have been studied previously for achieving this goal but these have met with limited success. Due to the radioactive nature of spent fuel, decay energy is being deposited in the fuel at a relatively constant rate. That decay energy leads to self-induced x-ray fluorescence of the uranium and plutonium atoms in the fuel. These resulting x-rays are then emitted by the fuel rod and can be measured in an appropriately designed and implemented instrument. The presence of uranium x-rays has been observed on numerous occasions; however, due to its dilute nature in the spent fuel and the presence of a large background, the plutonium x-rays have only been observed in a small number of experiments and generally with fuel containing very large loadings of plutonium. In this work, a feasibility study was conducted using both Monte Carlo simulations and measurements of PWR spent fuel rods at Oak Ridge National Laboratory as part of the Coupled End-to-End (CETE) demonstration. This feasibility study demonstrated the measurability of the plutonium x-rays for PWR spent fuel with burnups ranging from 35 to 70 GWd/MTU and the potential application of this technique as a quantitative assay tool.

Keywords: spent fuel; nondestructive analysis; x-ray fluorescence; plutonium quantification

1. Introduction

One of the most difficult non-destructive assay (NDA) problems in safeguards is quantitative measurements of plutonium (Pu) in spent nuclear fuel. Light water reactor spent fuel contains approximately 1% Pu and 3% fission products with the remainder uranium (U). The fission products produce an intense gamma-ray field that obscures the Pu gamma rays. The radioactive decay of the spent fuel however will induce fluorescence in the U and Pu and produce K x-rays. These x-rays might prove to be a useful NDA signature of the Pu to U ratio in the spent fuel.

The U and Pu K x-rays range from about 94 to 120 keV. The energy and relative intensities of these x-rays is shown in Table 1. Since the bulk of the spent fuel is U, the Pu x-rays with similar energy to the U x-rays will likely be obscured. However, the 103.7 keV $K_{\alpha 1}$ x-ray of Pu is relatively well separated from the U x-rays and may be measureable in spent fuel.

Table 1. Uranium and plutonium x-ray data [1].

X Ray	Energy (keV)		Relative Intensity	
	Uranium	Plutonium	Uranium	Plutonium
$K_{\alpha 1}$	98.44	103.76	100	100
$K_{\alpha 2}$	94.67	99.55	61.9	62.5
$K_{\beta 1}$	111.30	117.26	22.0	22.2
$K_{\beta 2}$	114.50	120.60	12.3	12.5
$K_{\beta 3}$	110.41	116.27	11.6	11.7

Measurement of U and Pu x-rays has been performed previously. This signature is routinely measured in small aliquots of spent fuel dissolutions [2]. However, the plutonium K x-rays have only rarely been measured in solid spent fuel. The first published observation of the 103.7 keV Pu x-ray from spent fuel was by C. Rudy et al. in 1998 [3]. This was a measurement of BN-350 fast breeder reactor spent fuel which had been cooling for 5-10 years prior to measurement. The ratio of the 103.7 keV peak area to the continuum was low and the peak areas had large errors, but the detector being used for this measurement was not specifically designed to measure these x-rays. Thus, it is expected that an optimized and collimated detector system might be capable of measuring this signal for spent fuel in a dry hot cell.

In this work, a study was performed to determine the feasibility of using self-induced x-ray fluorescence (XRF) as a direct measure of the Pu content of solid spent fuel pins in a dry hot cell. Experiments were performed on spent fuel rods at Oak Ridge National Laboratory (ORNL) to determine if a well designed measurement system could measure the 103.7 keV x-ray from Pu and if it could be correlated to Pu content in the fuel. Simulations were also performed to aid in detector design, data analysis, and fundamental understanding of this technique.

2. Spent Fuel Measurements

Three measurement campaigns were performed at ORNL in May 2008, July 2008, and January 2009. All measurements were performed in the hot cells at Building 3525 at ORNL and all measurements were for PWR spent fuel in individual rod segments. The measurements in May 2008 failed to produce a viable Pu x-ray measurement due to the large Compton continuum from gamma-ray interactions in the fuel, detector, shielding, and other materials. The detector arrangement was simulated with MCNPX [4] and these simulations suggested possible changes to the detector shielding which would decrease the Compton continuum by as much as a factor of 10. In July 2008, modifications to the detector shielding were performed based on the results from these simulations and this resulted in a measureable Pu x-ray signal. In January 2009, additional modifications to the detector and experimental arrangement were performed to provide an increased signal-to-noise ratio. Thus, Pu/U x-ray ratios were shown to be measureable for LWR spent fuels with burnups ranging from 30-70 GWd/tU. The description in the remainder of this paper is limited to the January 2009 measurement campaign which covere a burnup range of 25-50 GWd/tU.

2.1 Spent Fuel Characteristics

All measurements described here were performed on rod D5 from assembly NJ05YU from TMI-1. This rod had an average burnup of approximately 50 GWd/tU. The characteristics of the rod and assembly are given in Table 2. The rod had been previously cut into segments of varying lengths. These segments were then packaged into stainless steel shipping tubes with a thickness of approximately 0.16 cm. The rods were measured while still in the shipping tubes.

Table 2. Characteristics of TMI-1 spent fuel rod D5 from assembly NJ05YU [5].

Parameter		Value
Assembly and reactor data		
	Design	B&W PWR
	Lattice geometry	15x15
	Rod pitch (cm)	1.4427
	Number of fuel rods	208
	Number of guide tubes	16
	Number of instrumentation tubes	1
	Assembly pitch (cm)	21.8110
Fuel rod data		
	Fuel material	UO_2
	Fuel pellet density (g/cm^3)	10.196
	Fuel pellet diameter (cm)	0.9362
	Clad material	Zircaloy-4
	Clad inner diameter (cm)	0.9576
	Clad outer diameter (cm)	1.0922
Guide/instrument tube data		
	Guide/ instrumentation tube material	Zircaloy-4
	Guide tube inner diameter (cm)	1.2649
	Guide tube outer diameter (cm)	1.3462
	Instrumentation tube inner diameter (cm)	1.1201
	Instrumentation tube outer diameter (cm)	1.2522

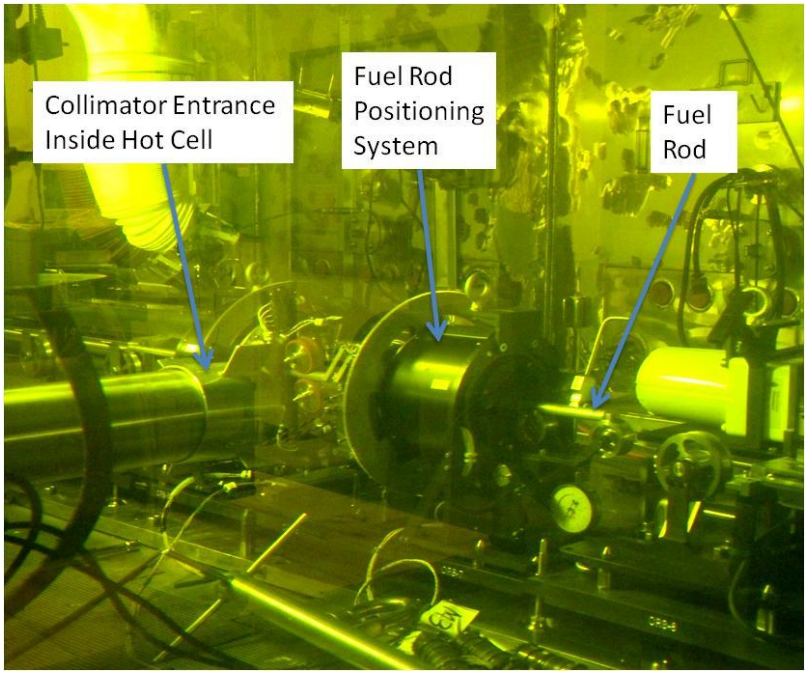

Figure 1. Collimator, fuel rod, and rod positioning system arrangement inside hot cell.

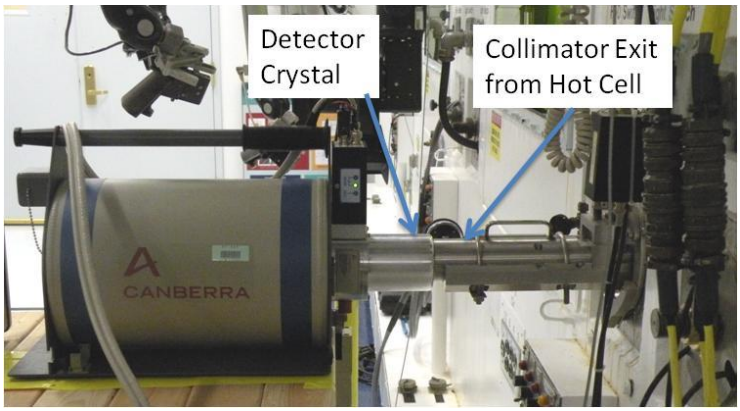

Figure 2. Detector and collimator arrangement outside hot cell.

2.2 Experimental Arrangement and Procedure

The spent fuel rod segments were placed in the hot cell in building 3525 at ORNL. This hot cell contains a rod positioning system (shown in Figure 1) that allowed the rod to be moved laterally in front of a collimator that would collimate gamma rays into a thin beam projecting through the hot cell wall. The collimator was 65.75" long and the hot cell walls were 36.0" thick. The collimator extended 27.5" into the hot cell. The fuel rod was 8" from the end of the collimator inside the hot cell. There was an extension on the collimator that extended 12.25" outside the hot cell wall. A Canberra Model GL0515R planar detector was placed directly against this extension (shown in Figure 2). The tungsten collimator on the detector was removed and no shielding was placed around the detector. Thus, the detector front face was 73.75" from the fuel rod. A similar set of measurements were also performed with a coaxial detector to acquire a broad energy spectrum for each location as well. Spectra were collected from various positions along the fuel rod with the majority of positions being near the top of the fuel rod (where the fuel burnup changed the most with change in distance from the end of the rod). Count times varied from 1-16 hours. Detector dead times ranged from 6-16% depending upon the rod position. With the detector and collimator in this arrangement and using fairly long count times, the 103.7 keV x-ray peak from Pu was clearly visible. An example spectrum is shown in Figure 3.

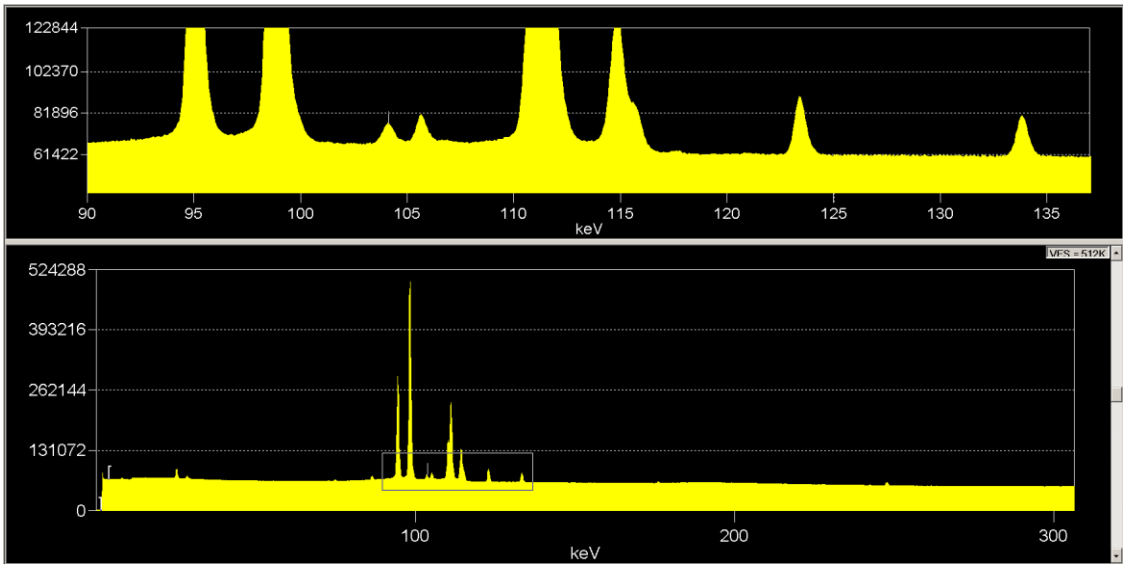

Figure 3. Measured spectra from TMI spent fuel rod D5 showing the Pu x-ray peak at 103.7 keV.

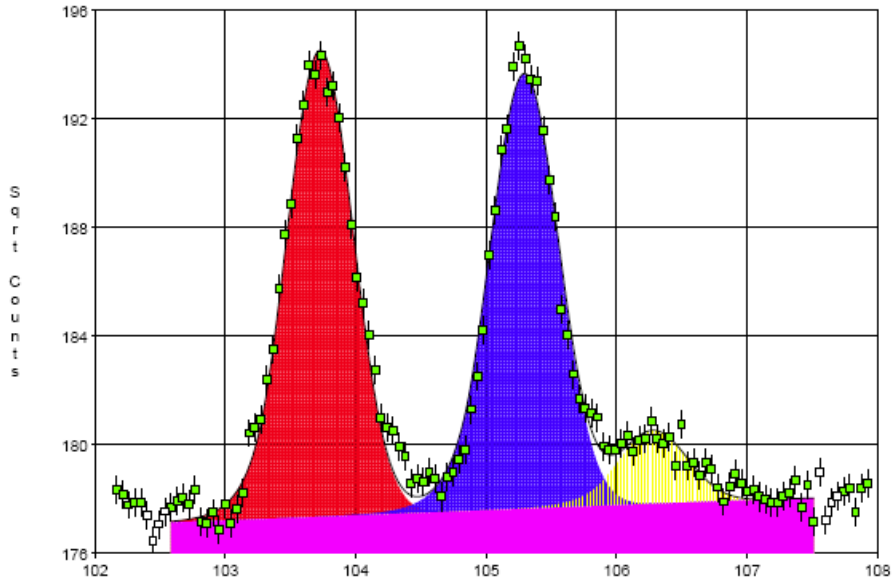

Figure 4. Example peak fit for Pu x-ray peak at 103.7 keV using the GENIE 2000 interactive peak fit.

2.3 Spectrum Analysis

The measured spectra were analyzed using Canberra's GENIE 2000 interactive peak fit. An example peak fit is shown in Figure 4. As can be seen, there is a peak at 105.3 keV from Eu-155 that is near to the Pu K x-ray peak, but with a high resolution detector, these peaks are easily separable. The spectra were analyzed to acquire count rates for the U K x-ray at 98.4 keV, the Pu K x-ray at 103.7 keV, the Cs-134 gamma ray at 604 keV, and the Cs-137 gamma ray at 661 keV. The ratio of Cs-134/Cs-137 is an indicator of fuel burnup [6]. The measured Pu/U K x-ray ratio was then plotted versus the measured Cs-134/Cs-137 gamma ray ratio (Figure 5). As can be seen, a strong correlation between Pu/U x-ray ratio and Cs-134/Cs-137 gamma ray ratio is found.

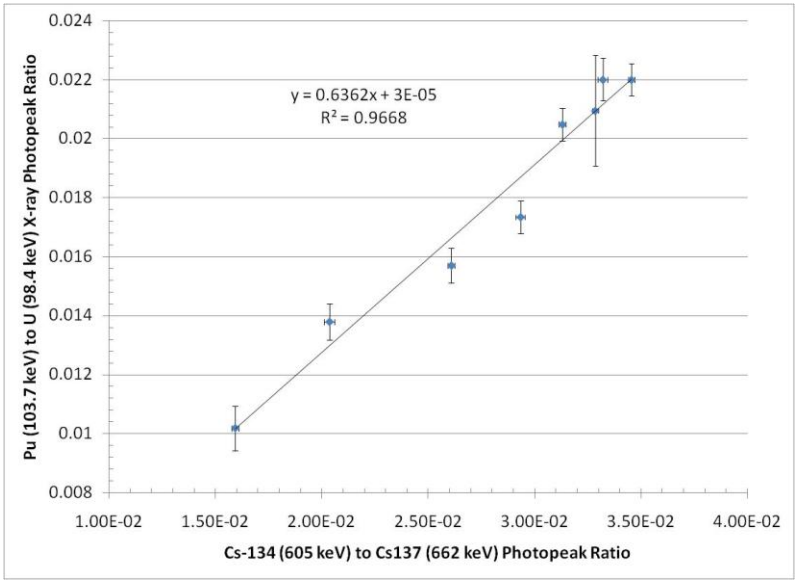

Figure 5. Measured Pu/U x-ray ratio versus measured Cs-134/Cs137 gamma ray ratio for TMI fuel rod D5.

3. Simulations and Results

Destructive analysis of the samples measured for rod D5 had not yet been performed at the time of writing this paper. Thus, simulations were used to relate the measured Cs-134/Cs-137 gamma-ray ratio to burnup and expected Pu/U content in the spent fuel to prove the feasibility of this technique. A two-dimensional TransLAT simulation [7] was used to determine the expected spent fuel isotopics as a function of fuel burnup and radial position in the fuel pin. This simulation included 20 radial fuel regions in the fuel pin. The fuel pin was burned from 0 to 70 GWd/tU and included the declared decay time for the TMI-1 D5 rod from time of discharge to time of measurement. The calculate average Pu/U atom ratio in the fuel pin versus Cs-134/Cs-137 atom ratio in the pin is shown in Figure 6.

When correlated to the measured Cs-134/Cs-137 ratio from Figure 5, we find that the Pu/U atom ratio predicted using Figure 6 is very low. For example a measured Cs-134/Cs-137 ratio of 0.025 would correspond to a measured Pu/U ratio of approximately 0.016 using Figure 5. However, the same Cs-134/Cs-137 ratio of 0.025 would correspond to a Pu/U ratio of 0.003 from Figure 6. It was discovered that this is because the measured Pu/U x-ray ratio is heavily influenced by the concentration of Pu on the outer surface of the fuel pin due to the strong attenuation of the 103.7 keV x-rays in the fuel.

Figure 7 shows the Pu concentration as a function of radial position in the fuel for several burnups. The U concentration only changes slightly as a function of position within the pin, but the Pu concentration is almost a factor of 5 higher on the outer surface of the pin than in the center of the pin. The Cs-134/Cs-137 atom ratio change only slightly (about 5-10%) as a function of radial position in the pin as well, and there is only minimum attenuation of the 605 and 661 keV gamma rays from Cs-134 and Cs-137 regardless of where in the pin they are produced in the pin. Thus, the measured Cs-134/Cs-137 gamma-ray ratio is a good indicator of the average Cs-134/Cs-137 atom ratio in the entire pin.

The TransLAT calculated Pu/U atom ratio on the surface of the fuel pin was correlated to the average Cs-134/Cs-137 atom ratio for the pin. This correlation was then used to infer the Pu/U atom ratio at the surface of the pin from the measured Cs-134/Cs-137 gamma-ray ratio. A plot of the measured Pu/U x-ray ratio versus the inferred Pu/U atom ratio at the surface of the pin is shown in Figure 8. This plot shows that the measured x-ray ratio is a good measure of the Pu/U concentration at the surface of the pin. The Pu/U concentration at the surface of the pin can be directly related to the average Pu/U ratio for the entire pin given knowledge of the pin dimensions (which we would expect would be known). Thus, it is feasible to measure the Pu/U x-ray ratio and Cs-134/Cs-137 gamma ray ratio for LWR spent fuel pins with burnups from 30-70 GWd/tU and use this data to determine the Pu/U atom ratio in the pin.

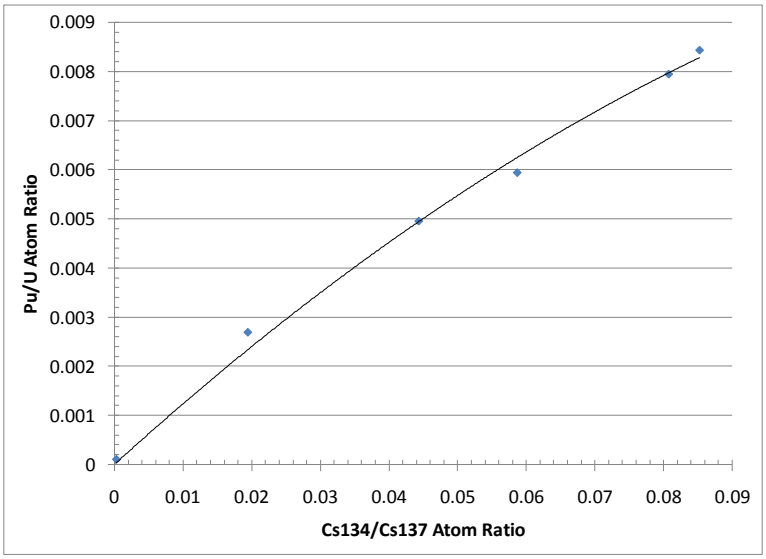

Figure 6. Calculated volumetrically averaged Pu/U atom ratio versus Cs-134/Cs-137 atom ratio for TMI fuel rod D5.

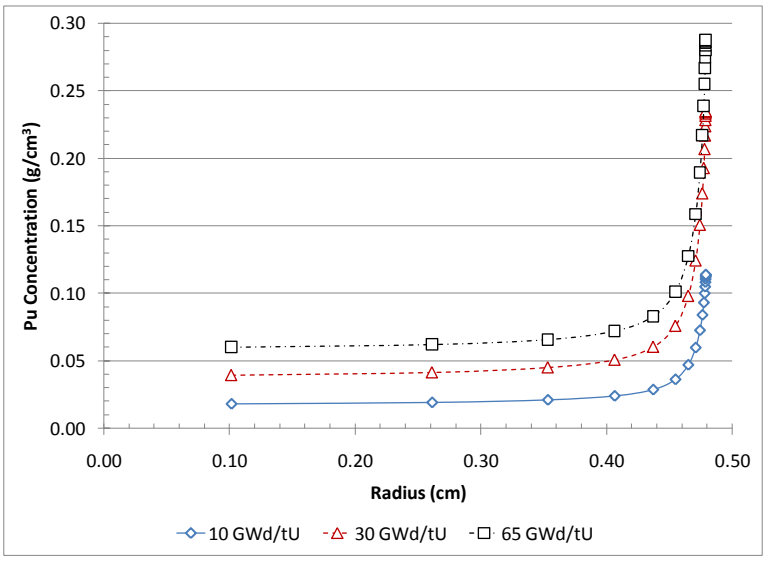

Figure 7. Calculated Pu concentration versus radius for TMI fuel rod D5.

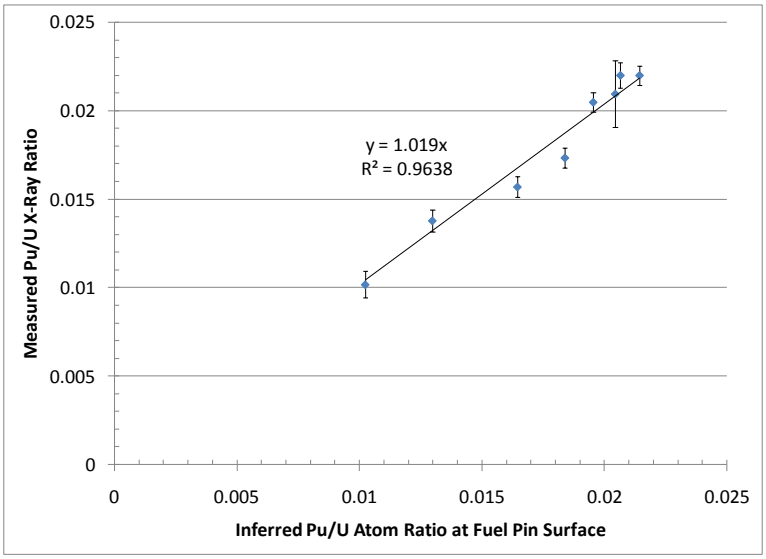

Figure 8. Measured Pu/U x-ray ratio versus Pu/U atom ratio at fuel pin surface inferred from the measured Cs-134/Cs-137 gamma-ray ratio.

4. Conclusion

Measurements and simulations were used to demonstrate that with a properly configured detector system, it is feasible to measure the Pu/U atom ratio at the surface of a fuel pin using measurements of the 103.7 keV K x-ray from Pu. Since the Pu/U atom ratio at the surface of the pin is directly related to the average Pu/U concentration for the pin, this implies that this technique will be feasible for measuring the Pu/U concentration in an LWR spent fuel pin. However, this requires knowledge of the fuel pin design and operating history to properly correlate surface Pu to average Pu for a pin. Also, the count times that were used in these measurements were longer than would be acceptable in most applications. These measurements were also restricted to individual fuel pins and it is ulikely that this technique could be

extrapolated to entire fuel assemblies. For future work, optimized detector systems and measurement approaches will be explored to develop a system that can measure the Pu/U x-ray ratio with shorter count times. Also, the sensitivity of this method to variations in system parameters will be studied to determine the expected accuracy of this technique.

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LA-UR 09-03574

Development of Self-Interrogation Neutron Resonance Densitometry (SINRD) to Measure the ^{235}U and ^{239}Pu Content in a PWR 17x17 Spent Fuel Assembly

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Abstract:

The use of Self-Interrogation Neutron Resonance Densitometry (SINRD) to measure the ^{235}U and ^{239}Pu content in a PWR spent fuel assembly was investigated via Monte Carlo N-Particle eXtended transport code (MCNPX) simulations. The sensitivity of SINRD is based on using the same fissile materials in the fission chambers as are present in the fuel because the effect of resonance absorption lines in the transmitted flux is amplified by the corresponding (n,f) reaction peaks in fission chamber. These simulations utilize the ^{244}Cm spontaneous fission neutrons to self-interrogate the fuel pins. The amount of resonance absorption of these neutrons in the fuel can be measured using ^{235}U and ^{239}Pu fission chambers placed adjacent to the assembly. We used ratios of different fission chambers to reduce the sensitivity of the measurements to extraneous material present in fuel. The development of SINRD to measure the fissile content in spent fuel is of great importance to the improvement of nuclear safeguards and material accountability. Future work includes the use of this technique to measure the fissile content in FBR spent fuel and heavy metal product from reprocessing methods.

Keywords: spent fuel, nuclear safeguards, fissile content, plutonium

1. Introduction

The development of non-destructive assay (NDA) capabilities to measure the fissile content in nuclear fuels is crucial to the implementation of effective international safeguards. The use of self-interrogation neutron resonance densitometry (SINRD) for the assay of fissile materials is a promising technique for nuclear safeguards and material accountability measurements. The neutron resonance cross-section structure is unique for each of the fissionable isotopes such as ^{235}U, ^{233}U, ^{239}Pu, and ^{241}Pu, and the resonance structure can provide a signature for the measurement of these materials of importance for safeguards and non-proliferation. The sensitivity of this technique is based on using the same fissile materials in the sample and fission chamber because the effect of resonance absorption lines in the transmitted flux is amplified by the corresponding (*n,f*) reaction peaks in the fission chamber. Thus, a ^{235}U fission chamber has high sensitivity to the neutron resonance absorption in ^{235}U that is in the sample, and similarly for the other fissile isotopes. The self-interrogation signature is a result of having the same fissile material in the fission chamber as in the sample [1].

In Fig. 1, the ^{239}Pu fission cross-section is compared to the resonance absorption lines in the neutron flux after transmission through a Gd filter and 0.25-mm [curve (a)] and 2.54-mm [curve (b)] ^{239}Pu metal sample. It is important to note that as the sample thickness increases, the self- interrogation signature decreases due to self-shielding effects occurring from saturation of the larger resonances [2].

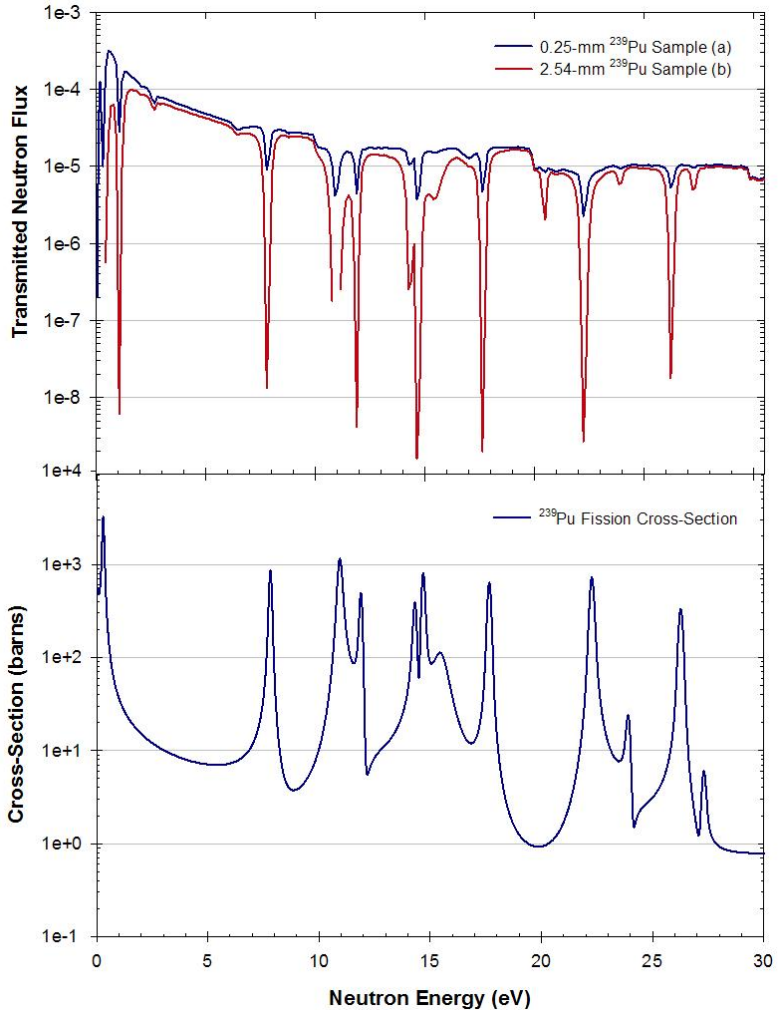


Figure 1. Comparison of absorption lines in the neutron flux after transmission through a 0.114-mm Gd filter and 0.25-mm [curve (a)] and 2.54-mm [curve (b)] ^{239}Pu metal sample (upper plot) to the ^{239}Pu fission cross-section at neutron energies ≤ 30 eV (bottom plot) [2].

The primary objective of this research is to develop and assess the sensitivity of using Self-Interrogation Neutron Resonance Densitometry (SINRD) for nuclear safeguards measurements. Recent interest in this approach was stimulated by an IAEA request related to spent fuel verification. Prior measurements [3,4] and calculations [1] have demonstrated that the SINRD method gives quantitative results for the fissile concentration in metal plates, MOX fuel rods, and a PWR 17x17 fresh fuel assembly [5]. The work described in this paper is focused on investigating the use of SINRD to measure the ^{235}U and ^{239}Pu content in PWR 17x17 spent LEU and spent MOX fuel assemblies via Monte Carlo N-Particle eXtended transport code (MCNPX) [6] simulations. The results from these simulations were used to optimize the detector configuration, assess the sensitivity and penetrability of SINRD to partial defects (i.e. missing fuel pins) and obtain a better understanding of the underlying physics of this measurement technique.

We varied the fuel burnup from 10-GWd to 50-GWd (in 10-GWd increments) to observe how the measured response changes as a function of ^{235}U and ^{239}Pu content in the fuel. SCALE 5.1 [7] was used to calculate the isotopic composition of PWR spent LEU and MOX fuel at each burnup step. It is important to note that in the MCNPX simulations, the spent fuel isotopics were assumed to be homogeneously distributed in the fuel pins. To assess the sensitivity and penetrability of the SINRD technique, we uniformly removed fuel pins from three different regions of the assembly assuming four-quadrant symmetry and replaced them with depleted uranium (DU) pins. The goal of this analysis is to calculate the percent change in the SINRD ratios per pin removed for each region to determine the minimum number of diverted rods that can be detected with a 2σ confidence level.

2. Description of Measurement System

PWR 17x17 spent LEU and MOX fuel assemblies were simulated in water (with and without 2200-ppm of boron) to determine how the scattering of neutrons in water affects the detector response. Spontaneous fission neutrons from ^{244}Cm were used to self-interrogate the spent fuel pins in the MCNPX simulations of SINRD. The concentration of ^{235}U and ^{239}Pu in the spent fuel pins was determined by measuring the distinctive resonance absorption lines from ^{235}U and ^{239}Pu using both ^{235}U and ^{239}Pu fission chambers (FC) placed adjacent to the side of the fuel assembly. Ratios of different fission chambers were used to reduce the sensitivity of the measurements to extraneous material present in fuel (e.g. fission products). This also reduces the number of unknowns we are trying measure because the neutron source strength and the detector-fuel assembly coupling cancels in the ratio. The specifications used to model the fuel assembly are given in Table 1.

Specifications		**PWR 17x17**
Assembly width (square)		212 mm
Lattice dimensions		17 x 17
Number of pins per assembly		264
Fuel material		UO_2 / MOX
Initial Fissile Content	LEU Fuel	4.0 at% ^{235}U
	MOX Fuel	4.0 wt% Pu
Cladding material		Zircaloy 2
Outer fuel diameter		9.00 mm
Outer clad diameter		10.0 mm
Fuel element pitch		12.5 mm
Moderator		Light Water

Table 1: Characteristics of PWR 17x17 spent fuel assembly.

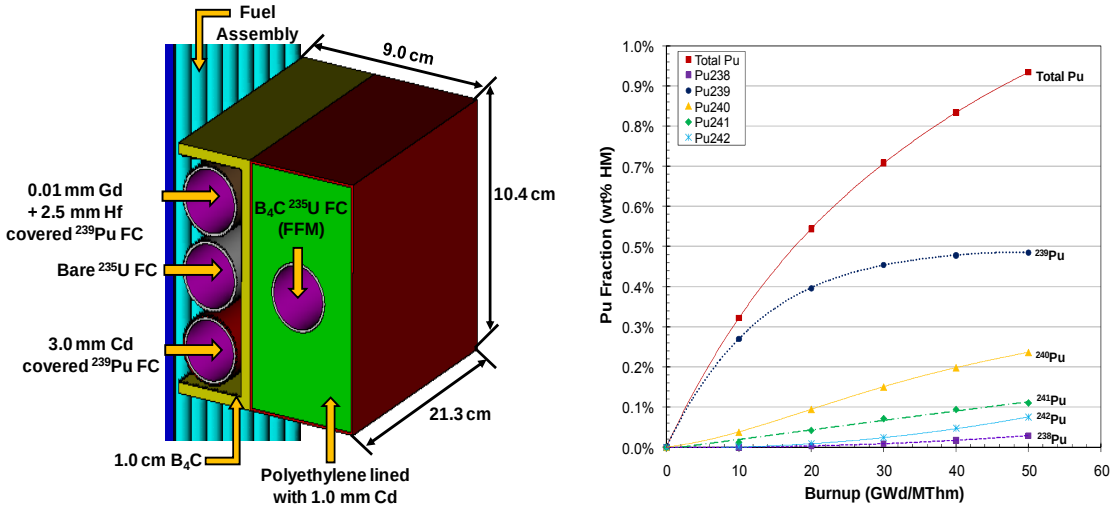
The SINRD detector unit is located adjacent to the assembly and is approximately 21.3-cm long, 10.4-cm high, and 9.4-cm wide. In order to reduce the background from thermal neutrons, the sides and back of the detector pod were covered with either 1.0-cm of boron carbide (B_4C) or 1.0-mm of Cd. The outer ^{235}U fission chamber (behind B_4C) was embedded in polyethylene to thermalize the fast neutrons that penetrated the boron shielding to increase counting statistics. The neutron flux entering the detector pod was measured using two fission chambers. The bare ^{235}U fission chamber was used to measure the entire neutron spectrum with thermal-neutron domination, and the outer ^{235}U fission chamber located behind the B_4C shield was used to monitor the fast neutron flux above neutron energies in the resonance region. The SINRD detector configuration was optimized for both PWR spent LEU and MOX fuel cases based on the different concentrations of ^{239}Pu relative to ^{240}Pu present in each case over the burnup range of 0 to 50 GWd.

3. Analysis of PWR Spent LEU Fuel

3.1. Optimization of SINRD Detector Ratios and Absorber Filters

The fission chamber ratios that can be used for SINRD consist of Gd+Hf and Cd covered ^{239}Pu FCs and two neutron flux monitors, Bare ^{235}U FC and B_4C ^{235}U FC (or Fast Flux Monitor). In this study, the sensitivity of the SINRD technique to different combinations of filters and monitors was investigated to determine the optimum configuration that maximized the detector ratio signature. Figure 2 shows the optimized detector configuration used to determine the ^{235}U and ^{239}Pu content in a PWR spent LEU fuel assembly and the in-growth of plutonium isotopics in spent LEU fuel as a function of burnup.

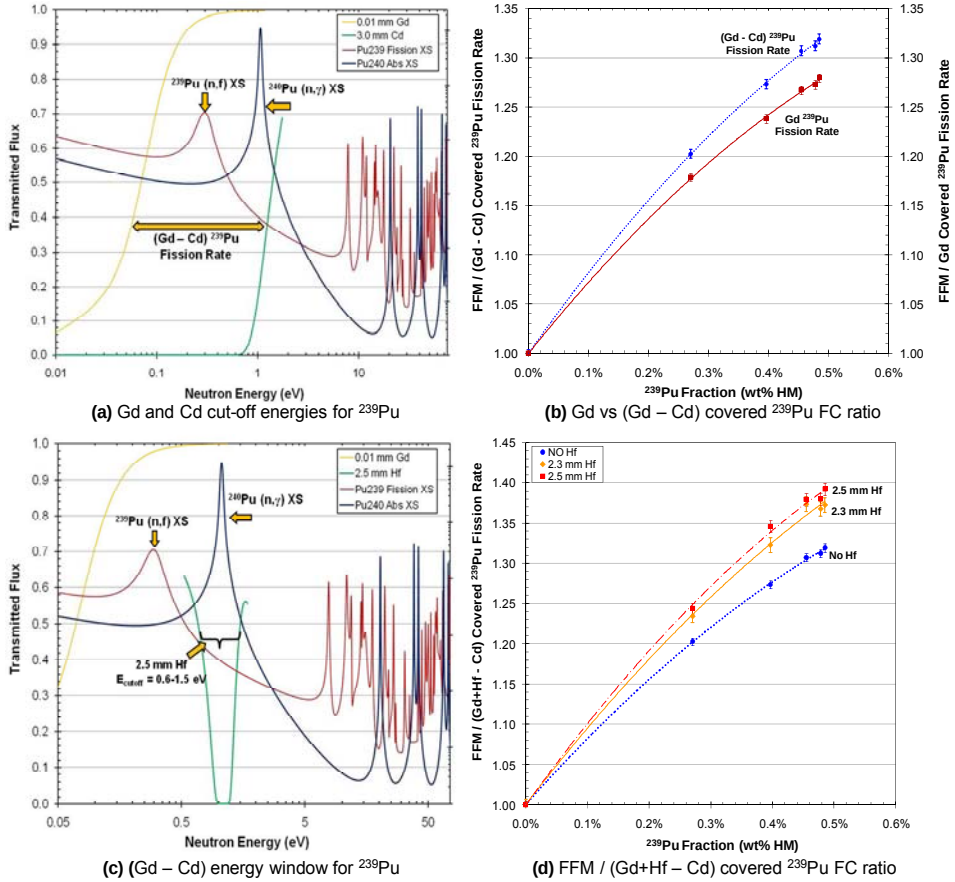
The Gd+Hf and Cd covered ^{239}Pu FCs are used to measure the resonance absorption from ^{239}Pu in the spent fuel. The transmitted flux through each of these filters relative to the ^{239}Pu (n,f) and ^{240}Pu (n,γ) cross-sections, as well as, the results from testing various combinations of these absorber filters to maximize the SINRD detector ratio signature for measuring ^{239}Pu are shown in Fig. 3. It should be noted that in the following results, we refer to the B_4C ^{235}U FC as Fast Flux Monitor (or FFM).



(a) SINRD Detector Configuration

(b) Plutonium Isotopics versus Burnup

Figure 2. (a) SINRD detector configuration used to determine ^{235}U and ^{239}Pu content in a PWR spent LEU fuel assembly and (b) plutonium isotopics in spent LEU fuel versus burnup.



(a) Gd and Cd cut-off energies for ^{239}Pu

(b) Gd vs (Gd – Cd) covered ^{239}Pu FC ratio

(c) (Gd – Cd) energy window for ^{239}Pu

(d) FFM / (Gd+Hf – Cd) covered ^{239}Pu FC ratio

Figure 3. Optimized detector ratios and filters for ^{239}Pu measurements: (a) the ^{239}Pu (n,f) and ^{240}Pu (n,γ) cross-sections within the (Gd – Cd) absorption cut-off energy window, (b) ratio of FFM / Gd covered ^{239}Pu FC compared to FFM / (Gd – Cd) covered ^{239}Pu FC ratio versus ^{239}Pu fraction (wt% HM), (c) transmitted flux through 2.5 mm Hf relative to^{240}Pu (n,γ) cross-section, and (d) the FFM / (Gd – Cd) covered ^{239}Pu FC ratio versus ^{239}Pu fraction (wt% HM) for different thicknesses of Hf.

Figure 3a shows how the large ^{239}Pu resonance at 0.3 eV can be windowed in energy by using the (Gd – Cd) ^{239}Pu fission rate based on the location of Gd and Cd absorption cut-off energies relative to the ^{239}Pu fission cross-section. The thick Cd filter (3.0 mm) absorbs the majority of neutrons in the low energy region of the ^{239}Pu resonance whereas the thin Gd filter (0.01 mm) transmits the majority of these lower energy neutrons. Figure 3b shows the comparison of the FFM / Gd covered ^{239}Pu FC ratio versus FFM / (Gd – Cd) covered ^{239}Pu FC ratio as a function of ^{239}Pu fraction present in the spent fuel. Using the (Gd – Cd) ^{239}Pu fission rate in the detector ratio, increased the SINRD signature as shown in Fig. 3c. It is also important to note the linearity of the curves shown in Fig. 3b indicates that the SINRD ratio is tracking the ^{239}Pu concentration in the spent fuel well. The total neutron rate measured in the Fast Flux Monitor (FFM) increases rapidly with the burnup as shown in Fig. 4a. It should be emphasized that the results have been normalized to the fresh fuel case (initial enrichment = 4% ^{235}U).

To determine if the absorption of low energy neutrons by ^{240}Pu was decreasing our detector ratio signature, we investigated the effect of adding a Hf filter inside the Gd filter. Figure 3c shows the transmitted flux through a 2.5 mm Hf filter relative to the ^{240}Pu radiative capture cross-section. The Hf filter absorbs the majority of neutrons in the same energy region as the ^{240}Pu capture resonance. Figure 3d shows the FFM / (Gd+Hf – Cd) covered ^{239}Pu FC ratio as a function of ^{239}Pu fraction present in the spent fuel for no Hf, 2.3 mm and 2.5 mm of Hf. The addition of 2.5 mm Hf to the Gd covered ^{239}Pu FC increased the SINRD signature by 5.6%.

3.2. Verification of Burnup

Next, we investigated the use of our SINRD detector ratios to verify the burnup of a PWR spent LEU fuel assembly. The ^{235}U and ^{244}Cm fraction (wt%HM) versus burnup is shown in Fig. 4a. In Fig. 4b and 4c, we show the normalized FFM fission rate and Gd covered ^{239}Pu to Bare ^{235}U fission rate ratio as a function of burnup for two possible diversion scenarios where the burnup is misdeclared low and misdeclared high, respectively.

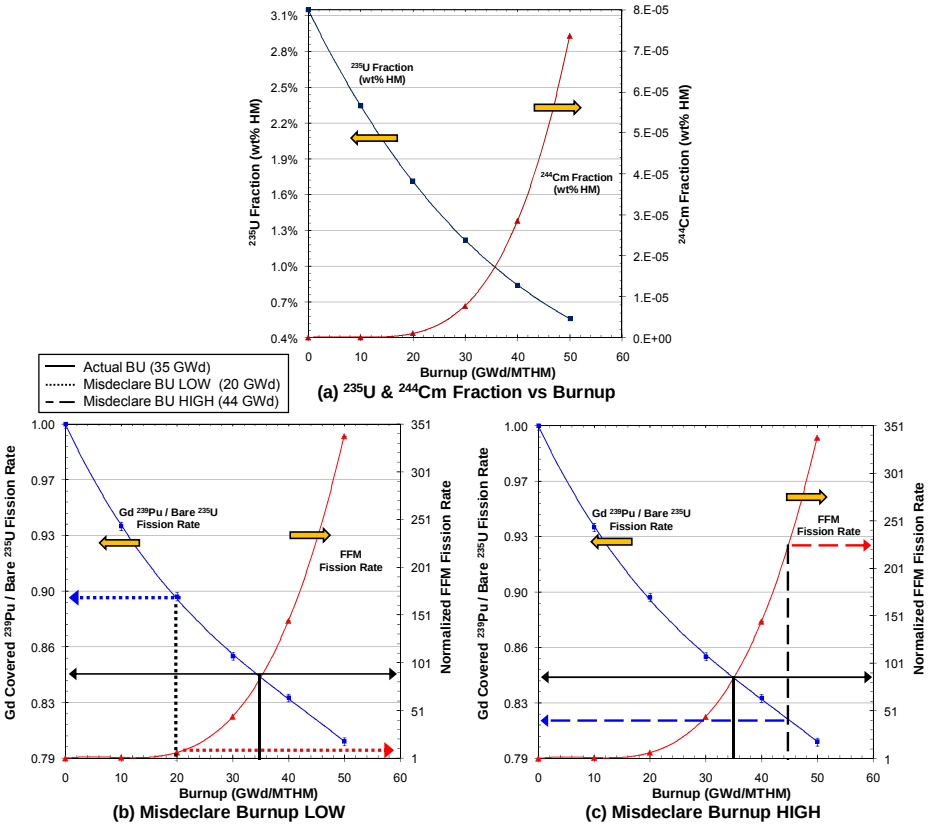


Figure 4. Comparison of (a) the ^{235}U and ^{244}Cm fraction (wt%HM) versus burnup to (b) and (c) which show the normalized FFM ratio and Gd covered ^{239}Pu FC / Bare ^{235}U FC ratio versus burnup for possible diversion scenarios where burnup is misdeclared low and misdeclared high, respectively.

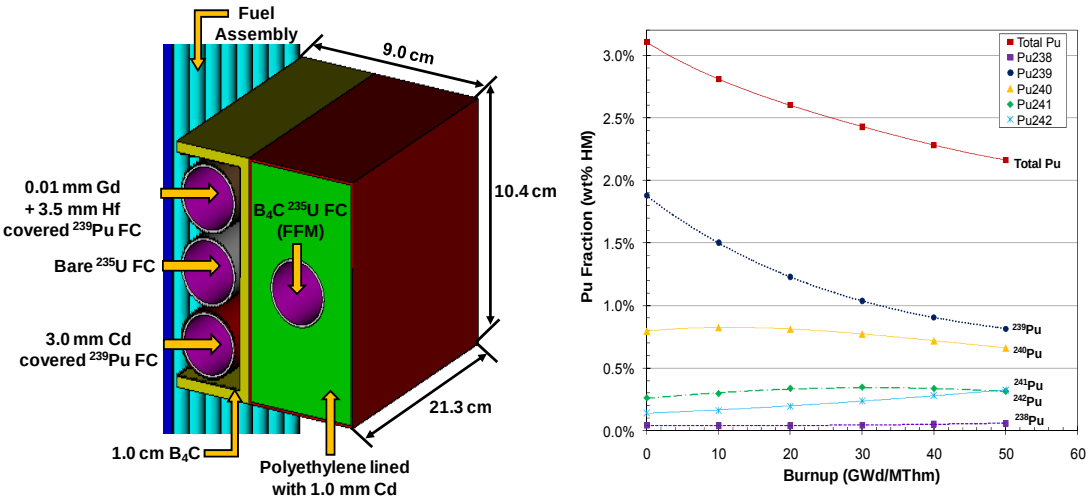
Comparison of Fig. 4a to Fig. 4b and 4c, clearly shows that the normalized fission rate in our FFM is accurately measuring the ^{244}Cm fraction and that the Gd covered ^{239}Pu to Bare ^{235}U fission rate ratio is accurately measuring the ^{235}U fraction over the burnup range of 0 – 50 GWd. The fact that ^{235}U fraction decreases as a function of burnup, whereas the ^{244}Cm fraction increases enables us to verify the burnup of the PWR spent LEU assembly because the proliferator can only get one of these curves right. For instance in Fig. 4b, we show the case where the burnup is misdeclared low. The solid black line indicates the actual burnup of the assembly which is 35 GWd and the solid black arrows point to the expected measured values at this burnup. The misdeclared burnup (20 GWd) is shown by the black dotted line and the dotted red and blue lines correspond to the expected measured values at the lower burnup. It should be noted that when the burnup is misdeclared the expected measured values move in opposite directions. Thus, comparing a set of measurements where the burnup is misdeclared to a reference measurement where the burnup is known would clearly indicate an anomaly in the declaration.

It should also be emphasized that the ^{244}Cm neutron emission rate from a PWR 17x17 spent LEU fuel assembly is approximately 1.0E+08 n/s and is further amplified by a factor of 2 – 3 by neutron multiplication in the water. This high neutron source term provides adequate counting statistics in the fission chambers to give better than 1% precision in a few minutes for the ratios.

4. Analysis of PWR Spent MOX Fuel Assembly

4.1. Optimization of SINRD Detector Ratios

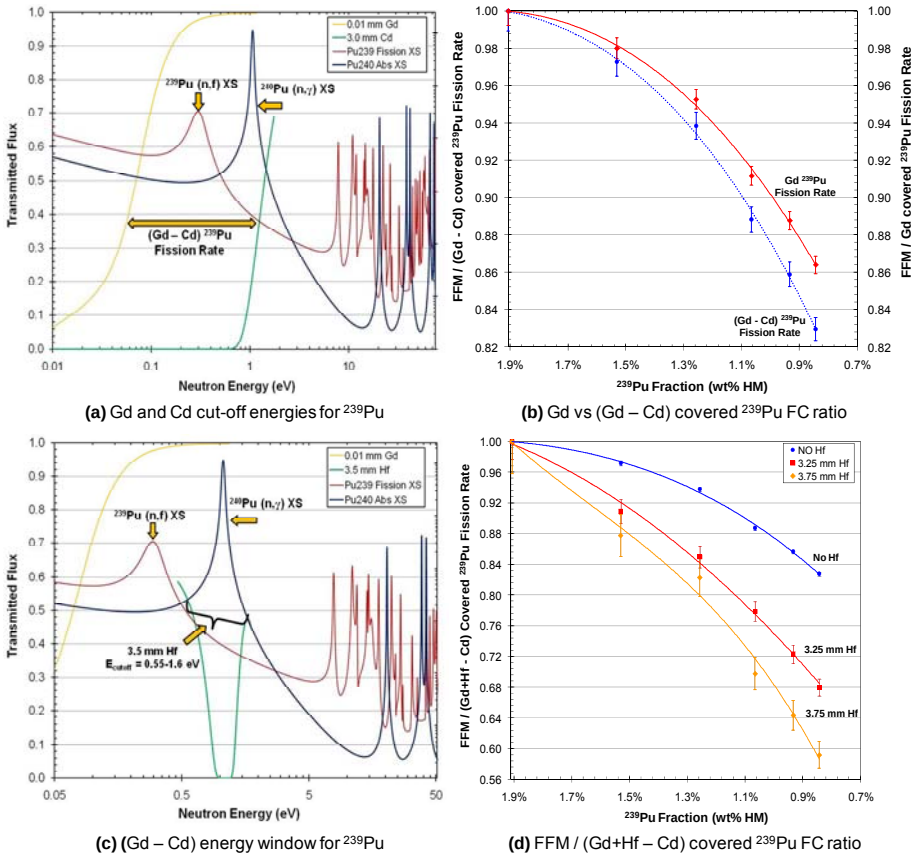
In order to better understand the physics of the SINRD technique, we have also simulated the use of SINRD to measure ^{239}Pu content in a PWR spent MOX fuel assembly. We believe that SINRD technique will work better for a PWR assembly with spent MOX fuel because the ^{239}Pu concentration is significantly larger and the ^{235}U concentration is significantly smaller (< 0.15 wt%HM) compared to PWR spent LEU fuel. Figure 5 shows the optimized detector configuration used to determine the ^{239}Pu content in a PWR spent MOX fuel assembly and the in-growth of plutonium isotopics in spent MOX fuel as a function of burnup. Since the ^{235}U fraction is less than 0.15 wt% heavy metal in PWR spent MOX fuel, we did not try to measure it. Once again, we have referred to the B_4C ^{235}U FC as FFM in the results presented in this section.



(a) SINRD Detector Configuration **(b)** Plutonium Isotopics versus Burnup

Figure 5. (a) SINRD detector configuration used to determine ^{239}Pu content in a PWR spent MOX fuel assembly and (b) plutonium isotopics in spent MOX fuel versus burnup.

Figure 6 shows the transmitted flux through Gd, Cd, and Hf absorber filters relative to the ^{239}Pu (n,f) and ^{240}Pu (n,γ) cross-sections and the results from testing various combinations of these absorber filters to maximize the SINRD detector ratio signature for measuring ^{239}Pu. It should be emphasized that the results have been normalized to the fresh fuel case (initial enrichment of MOX fuel = 4% Pu).



(a) Gd and Cd cut-off energies for ^{239}Pu

(b) Gd vs (Gd – Cd) covered ^{239}Pu FC ratio

(c) (Gd – Cd) energy window for ^{239}Pu

(d) FFM / (Gd+Hf – Cd) covered ^{239}Pu FC ratio

Figure 6. Optimized detector ratios and filters for ^{239}Pu measurements: (a) the ^{239}Pu (*n,f*) and ^{240}Pu (*n,γ*) cross-sections within the (Gd – Cd) absorption cut-off energy window, (b) ratio of FFM to Gd covered ^{239}Pu FC compared to FFM to (Gd – Cd) covered ^{239}Pu FC ratio versus ^{239}Pu fraction (wt% HM), (c) transmitted flux through 3.5 mm Hf relative to ^{240}Pu (*n,γ*) cross-section, and (d) the FFM / (Gd – Cd) covered ^{239}Pu FC ratio versus ^{239}Pu fraction (wt% HM) for different thicknesses of Hf.

Similar to Fig. 3a for PWR spent LEU fuel, we show how the large ^{239}Pu resonance at 0.3 eV can be windowed in energy by using the (Gd – Cd) ^{239}Pu fission rate in Fig. 6a. Figure 6b shows the ratio of the FFM / Gd covered ^{239}Pu FC ratio compared to the FFM to (Gd – Cd) covered ^{239}Pu FC ratio as a function of ^{239}Pu fraction present in the spent MOX fuel. Using the (Gd – Cd) ^{239}Pu fission rate in the detector ratio increased the SINRD signature by 4%.

Based on the fact that the ^{240}Pu concentration is much larger in PWR spent MOX fuel compared to PWR spent LEU fuel, we hypothesized that decrease in our detector ratio signature from the parasitic absorption of low energy neutrons by ^{240}Pu would be larger for spent MOX fuel. Thus, in order to offset this effect, we would have to increase the thickness of Hf. Figure 6c shows the transmitted flux through a 3.5 mm Hf filter relative to the ^{240}Pu radiative capture cross-section. Figure 6d shows the FFM / (Gd+Hf – Cd) covered ^{239}Pu FC ratio as a function of ^{239}Pu fraction present in the spent fuel for no Hf, 3.25 mm and 3.75 mm of Hf. The addition of 3.75 mm Hf to the Gd covered ^{239}Pu FC increased the SINRD signature by 32%. It should also be noted that the curves shown in Fig. 6d become more linear as the thickness of Hf is increased. This is important because it indicates that addition of Hf to our SINRD ratio enables us to more accurately track the ^{239}Pu concentration in the spent MOX fuel.

4.2. Verification of Burnup

Next, we investigated the use of SINRD to verify the burnup of a PWR spent MOX fuel assembly. The ^{239}Pu and ^{244}Cm fraction (wt%HM) versus burnup is shown in Fig. 7a. In Fig. 7b and 7c, we show the normalized FFM fission rate and the FFM / (Gd+Hf – Cd) covered ^{239}Pu fission rate ratio as a function of burnup for two possible diversion scenarios where the burnup is misdeclared low and misdeclared high, respectively.

Comparison of Fig. 7a to Fig. 7b and 7c, clearly shows that the normalized FFM fission rate is accurately measuring the ^{244}Cm fraction and that the FFM to (Gd+Hf – Cd) covered ^{239}Pu fission rate ratio is accurately measuring the ^{239}Pu fraction over the burnup range of 0 – 50 GWd. Similar to PWR spent LEU case, our ability to verify the burnup of the assembly is based on the fact that ^{239}Pu fraction decreases, whereas the ^{244}Cm fraction increases as a function of burnup. Thus, a proliferator who misdeclared the burnup of the assembly could only get one of these curves right because the expected measured values move in opposite directions.

In a PWR 17x17 spent MOX fuel assembly, the ^{244}Cm neutron emission rate is approximately 5.0E+08 n/s (~5x greater than a PWR spent LEU fuel assembly) and is further amplified by a factor of 2 – 3 by neutron multiplication in the water. This higher neutron source term in spent MOX fuel enables adequate counting statistics in the fission chambers to better than 1% precision for the ratios to be achieved in less than half the time that is required for spent LEU fuel.

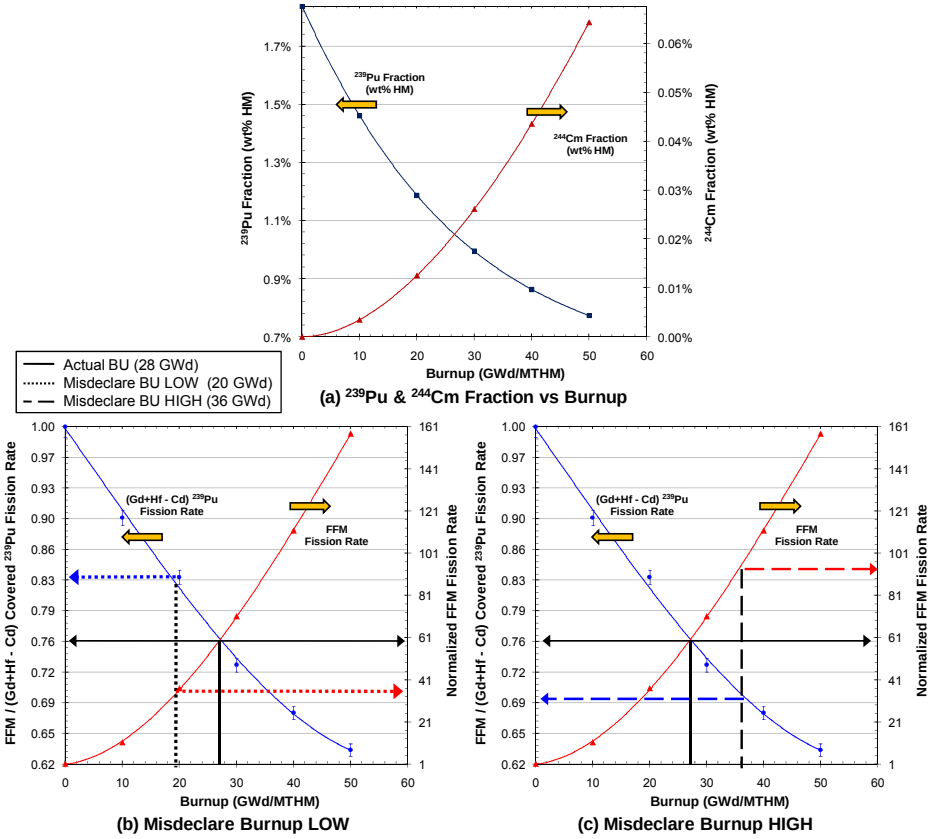


Figure 7. Comparison of (a) the ^{239}Pu and ^{244}Cm fraction (wt%HM) versus burnup to (b) and (c) which show the normalized FFM ratio and FFM / (Gd+Hf – Cd) covered ^{239}Pu FC ratio versus burnup for diversion scenarios where burnup is misdeclared low and misdeclared high, respectively.

5. Sensitivity of SINRD to Partial Defects

In general, there are two different models for the diversion of fissile material from a fuel assembly. The first is to misdeclare the burnup of the assembly, and the second is to remove fuel pins and to replace them with depleted uranium or iron pins. In the first model, the fissile material distribution is the same as for the calibration standard; however, for the second diversion model, the location of the pin diversion will affect the measured response based on the penetrability of the measurement technique.

Since the fission detector package can be applied to any of the four sides of the assembly, four-quadrant symmetry was assumed in the fuel loading and fuel removal. The penetrability of the SINRD technique was assessed by uniformly removing fuel pins from three different regions of the assembly

where Region 1 consists of two rows on the outer surface of the assembly, Region 2 consists of rows in the mid region, and Region 3 consists of rows in the center of the assembly. The pin removal locations of defects for Regions 1 - 3 in PWR 17x17 fuel assembly is shown in Fig. 8.

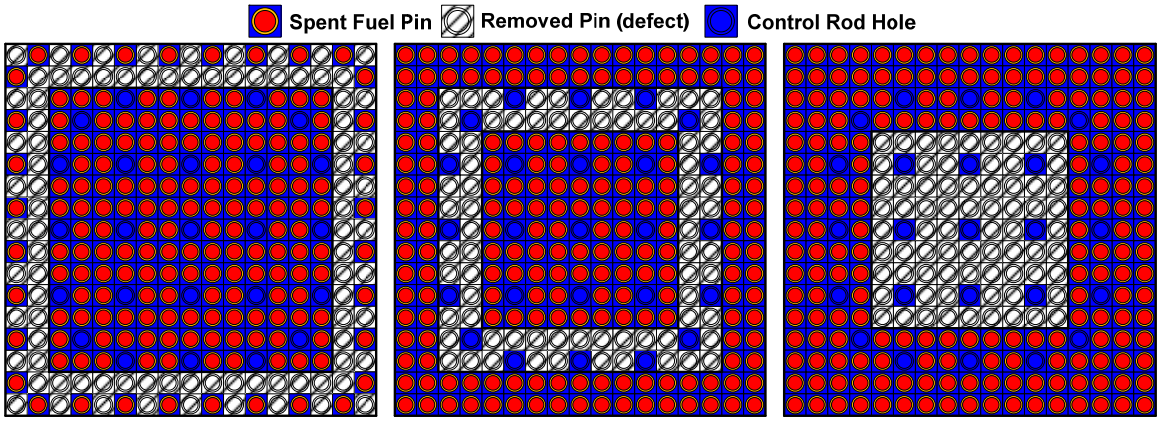


Figure 8. Pin removal locations of defects for Regions 1 (a), 2 (b) and 3 (c) in PWR 17x17 fuel assembly where the white pin locations represent the pins that were removed, and the blue locations are the control rods.

In Table 2, the sensitivity of six different SINRD detector ratios with 6.25%, 15%, and 27% of the total number of pins in the PWR spent fuel assembly removed from Regions 1, 2, and 3 for both spent LEU and spent MOX fuel (burnup = 40 GWd) is given.

Pin Defects (%)	Detector Ratio (NO boron)	Region 1 (1.25 cm) % Change in Ratio		Region 2 (3.75 cm) % Change in Ratio		Region 3 (7.75 cm) % Change in Ratio	
		LEU Fuel	MOX Fuel	LEU Fuel	MOX Fuel	LEU Fuel	MOX Fuel
6.25% Pin Defects	FFM / Bare ^{235}U	**8.31%**	**9.26%**	**3.67%**	**3.66%**	-1.30%	-1.15%
	FFM / Gd-Cd) ^{239}Pu	7.39%	8.91%	2.15%	2.63%	**-2.07%**	**-1.70%**
	FFM / Gd ^{239}Pu	6.77%	8.01%	1.82%	2.12%	-1.92%	-1.55%
	FFM / Cd ^{239}Pu	2.30%	2.45%	**0.02%**	**-0.03%**	-1.18%	-0.98%
	Bare ^{235}U/Gd ^{239}Pu	-2.76%	-2.47%	-2.20%	-1.82%	-0.59%	-0.38%
	Bare ^{235}U/Cd ^{239}Pu	**-10.8%**	**-13.5%**	**-4.34%**	**-4.38%**	**0.11%**	**0.15%**
15% Pin Defects	FFM / Bare ^{235}U	**19.9%**	**22.2%**	**8.80%**	**8.78%**	-3.12%	-2.75%
	FFM / Gd-Cd) ^{239}Pu	17.7%	21.4%	5.16%	6.32%	**-4.98%**	**-4.09%**
	FFM / Gd ^{239}Pu	16.2%	19.2%	4.37%	5.10%	-4.61%	-3.71%
	FFM / Cd ^{239}Pu	5.51%	5.87%	**0.06%**	**-0.06%**	-2.83%	-2.36%
	Bare ^{235}U/Gd ^{239}Pu	-6.63%	-5.93%	-5.27%	-4.38%	-1.41%	-0.92%
	Bare ^{235}U/Cd ^{239}Pu	**-25.9%**	**-32.3%**	**-10.4%**	**-10.5%**	**0.27%**	**0.37%**
27% Pin Defects	FFM / Bare ^{235}U	**36.3%**	**40.4%**	**16.0%**	**16.0%**	-5.68%	-5.00%
	FFM / Gd-Cd) ^{239}Pu	32.3%	38.9%	9.39%	11.5%	**-9.06%**	**-7.44%**
	FFM / Gd ^{239}Pu	29.6%	35.0%	7.95%	9.27%	-8.38%	-6.76%
	FFM / Cd ^{239}Pu	10.0%	10.7%	**0.11%**	**-0.11%**	-5.16%	-4.30%
	Bare ^{235}U/Gd ^{239}Pu	-12.1%	-10.8%	-9.60%	-7.97%	-2.56%	-1.67%
	Bare ^{235}U/Cd ^{239}Pu	**-47.1%**	**-58.8%**	**-18.9%**	**-19.1%**	**0.49%**	**0.67%**

Table 2: Sensitivity of six different SINRD detector ratios with 6.25%, 15%, and 27% partial defects removed from Regions 1, 2, and 3 for both spent LEU and spent MOX fuel (burnup = 40 GWd).

The highlighted values shown in Table 2 correspond to the MAXIMUM positive and negative percent change in ratios for **6.25%**, **15%**, **27%** partial defects removed from each region. To help distinguish the different fuel types, slightly darker shades of these colors were used for PWR spent MOX fuel. For

PWR spent LEU fuel, a count time of 500 seconds was used, whereas, a count time of 250 seconds was used for PWR spent MOX fuel. These different count times reflect the different concentrations of ^{244}Cm in spent LEU fuel versus spent MOX fuel. It is also important to emphasize that the percent change in detector ratio was greater than 2σ uncertainty for ALL ratios in ALL regions except for the Bare ^{235}U FC to Cd covered ^{239}Pu FC ratio for PWR spent LEU in Region 3 (cell has been shaded gray).

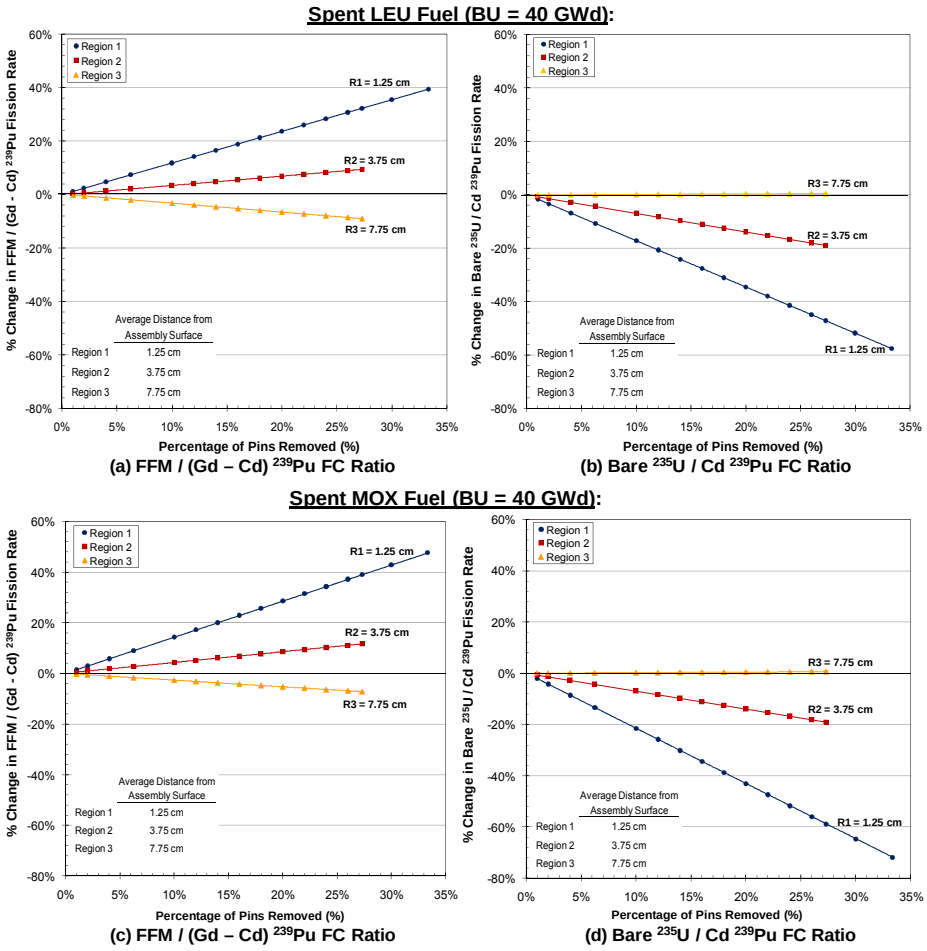


Figure 9. Sensitivity to partial defects: (a) and (c) % change in FFM / (Gd-Cd) covered ^{239}Pu fission rate ratio versus percentage of pins removed for *Spent LEU Fuel* and *Spent MOX Fuel*, respectively (BU = 40 GWd), (b) and (d) % change in Bare ^{235}U / Cd covered ^{239}Pu fission rate ratio versus percentage of pins removed for *Spent LEU Fuel* and *Spent MOX Fuel*, respectively (BU = 40 GWd).

The sensitivity of two different SINRD detector ratios to partial defects for PWR spent LEU fuel and spent MOX fuel at a burnup of 40 GWd is shown in Fig. 9. Figures 9a and 9c show the percent change in FFM / (Gd-Cd) covered ^{239}Pu fission rate ratio as a function of percentage of pins removed for *Spent LEU Fuel* and *Spent MOX Fuel*, respectively. In both cases, the sensitivity to partial defects is highest in Region 1 (two rows on the outer surface of the assembly). Based on these results (9a and 9c), it should be noted that there exists a combination of pins from Region 2 and Region 3 that could result in 0% percent change in FFM / (Gd-Cd) covered ^{239}Pu FC ratio.

Figures 9b and 9d show the percent change in the Bare ^{235}U / Cd covered ^{239}Pu fission rate ratio versus percentage of pins removed for *Spent LEU Fuel* and *Spent MOX Fuel*, respectively. Similar to the results shown in Fig. 9a and 9c, the sensitivity to partial defects is highest in Region 1 for both cases; however, the results shown in 9b and 9d go in the opposite direction as the results shown in 9a and 9c. Thus, the percent change in the Bare ^{235}U / Cd covered ^{239}Pu fission rate ratio could be used in conjunction with the percent change in FFM / (Gd-Cd) covered ^{239}Pu fission rate ratio such that

there is no combination of pins from Regions 2 and 3 that could result in a 0% percent change in the detector ratio.

For each case, error propagations (see Apendix A) were used to calculate the resulting uncertainties in the percent change in the ratio of the FFM / (Gd – Cd) covered ^{239}Pu FC and in the percent change in the ratio of the Bare ^{235}U FC / Cd covered ^{239}Pu FC. The uncertainties in these ratios were between 0.5% – 0.8% for count times of 500 seconds and 250 seconds for PWR spent LEU fuel and spent MOX fuel, respectively. Thus, this type measurement could show the departure from a reference fuel assembly with no defects. It should be emphasized that in all fuel assembly measurements, a reference assembly for calibration is assumed.

6. Conclusions

We have simulated the change in different SINRD detector ratios over a burnup range of 0 – 50 GWd using MCNPX. Most of the SINRD FC ratios have the Fast Flux Monitor rate (FFM) in the numerator where the FFM (or B_4C ^{235}U FC) is simply a fast-neutron flux monitor that measures the neutron source emission rate. The FFM / (Gd+Hf – Cd) covered ^{239}Pu fission rate ratio is sensitive to the ^{239}Pu content in both PWR spent LEU fuel and spent MOX fuel assemblies. The SINRD signature for ^{239}Pu concentration has not saturated for the ^{239}Pu fraction present in both cases over the burnup range of 0 to 50 GWd; however, the ^{239}Pu concentration in a PWR spent LEU fuel assembly is approaching saturation at a burnup of 50 GWd. For a factor of two change in the ^{239}Pu concentration, the signature ratio changes by 21% in water. Therefore, the sensitivity of the method to partial defects is limited to significant (> 10%) changes in the ^{239}Pu linear loading. This densitometry method requires a calibration with a reference assembly of similar geometry. However, the SINRD method uses the ratio of the FC detectors, so most of the systematic errors related to calibration and positioning cancel in the ratios.

The purpose of this paper was to study the SINRD method for PWR 17x17 spent LEU and spent MOX fuel assemblies. For the cases simulated in this paper, the spent fuel pins in the assembly present an approximate uniform sample to the transmitted neutrons because the self-shielding is small for individual pins and the Pu concentration was assumed to be homogeneously distributed in the pins. For spent fuel assemblies, the initial ^{235}U enrichment (LEU fuel) or initial Pu loading (MOX fuel) is tailored for the pin positions so that the in-growth of Pu (and burnup) is similar for the different pin positions. For the normal application of neutron (or gamma-ray) densitometry techniques, the sample is assumed to be homogeneous so that the transmitted beam provides the average concentration of the isotope of interest. This homogeneity is the normal condition for solutions and bulk powder, but not for fuel assemblies.

Future work will look at using SINRD to measure the fissile content in LWR spent fuel assemblies without the use of the Pu fission chambers. By using ^{235}U fission chambers plus metal foils to filter the neutron energy, we can still focus the measurement on the low-energy Pu resonances. The ratio of ^{235}U fission chambers with selected foil filters provide neutron energy spectral information that can be used to “fingerprint” the actinide loadings in spent fuel assemblies. In addition, the measurement of the ^{235}U and the ^{244}Cm can verify the burnup so that the burnup codes can provide the Pu isotopic ratios.

7. Acknowledgements

We would like to acknowledge the Program of Technical Assistance to the IAEA (POTAS) and the Department of Energy National Nuclear Security Administration’s Office of Global Security Engagement and Cooperation (NA-242) for their support in the development of the SINRD method. The IAEA has provided useful guidance and support for the activity.

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APPENDIX A

Count Rate in Detector

The count rate in detector (*i*) was calculated using the following equation where the subscript *i* = *B_4C ^{235}U, Bare ^{235}U, Gd+Hf covered,* or *Cd covered ^{239}Pu fission chambers* corresponding to the particular detector on which fission rate was tallied in MCNPX. The superscript *k* = *^{238}U or ^{240}Pu* corresponding to the spontaneous fission source in fresh LEU fuel or fresh MOX fuel, respectively.

$$CR_i = m_{SF}^k \cdot y_{SF}^k \cdot MCNPX\ Tally_i \tag{A.1}$$

where

$m_{SF}^k\ [g] \equiv$ Mass of the self-interrogating spontaneous fission source (*k*) in fuel

$y_{SF}^k\ [n/s \cdot g] \equiv$ Spontaneous fission yield of the self-interrogating spontaneous source (*k*) in fuel

$MCNPX\ Tally_i\ [fissions/source\ neutron] \equiv$ Fission rate tally in detector (*i*) from MCNPX output

Assuming a count time, t_C, the total number of counts in detector (*i*) and the corresponding standard deviation the counts were calculated using the following equations:

$$\begin{aligned} C_i &= CR_i \cdot t_C = m_{SF}^k \cdot y_{SF}^k \cdot MCNPX\ Tally_i \cdot t_C \\ \sigma_i &= \sqrt{C_i} \end{aligned} \tag{A.2}$$

Using the total number of counts calculated for each detector, six different detector ratios and corresponding standard deviations were calculated from the following equations (A.3) – (A.8):

$$1)\ R_1 = \frac{C_{B_4C}}{C_{Bare}},\ \sigma_{R1} = \frac{C_{B_4C}}{C_{Bare}}\sqrt{\left(\frac{\sigma_{B_4C}}{C_{B_4C}}\right)^2 + \left(\frac{\sigma_{Bare}}{C_{Bare}}\right)^2} \tag{A.3}$$

$$2)\ R_2 = \frac{C_{B_4C}}{C_{Gd-Cd}},\ \sigma_{R2} = \frac{C_{B_4C}}{C_{Gd-Cd}}\sqrt{\left(\frac{\sigma_{B_4C}}{C_{B_4C}}\right)^2 + \left(\frac{\sigma_{Gd-Cd}}{C_{Gd-Cd}}\right)^2} \tag{A.4}$$

$$3)\ R_3 = \frac{C_{B_4C}}{C_{Gd}},\ \sigma_{R3} = \frac{C_{B_4C}}{C_{Gd}}\sqrt{\left(\frac{\sigma_{B_4C}}{C_{B_4C}}\right)^2 + \left(\frac{\sigma_{Gd}}{C_{Gd}}\right)^2} \tag{A.5}$$

$$4)\ R_4 = \frac{C_{B_4C}}{C_{Cd}},\ \sigma_{R4} = \frac{C_{B_4C}}{C_{Cd}}\sqrt{\left(\frac{\sigma_{B_4C}}{C_{B_4C}}\right)^2 + \left(\frac{\sigma_{Cd}}{C_{Cd}}\right)^2} \tag{A.6}$$

$$5)\ R_5 = \frac{C_{Bare}}{C_{Gd}},\ \sigma_{R5} = \frac{C_{Bare}}{C_{Gd}}\sqrt{\left(\frac{\sigma_{Bare}}{C_{Bare}}\right)^2 + \left(\frac{\sigma_{Gd}}{C_{Gd}}\right)^2} \tag{A.7}$$

$$6)\ R_6 = \frac{C_{Bare}}{C_{Cd}},\ \sigma_{R6} = \frac{C_{Bare}}{C_{Cd}}\sqrt{\left(\frac{\sigma_{Bare}}{C_{Bare}}\right)^2 + \left(\frac{\sigma_{Cd}}{C_{Cd}}\right)^2} \tag{A.8}$$

Sensitivity to Partial Defects

Next, fuel rods were uniformly removed from Regions (1), (2) and (3) of the assembly and the six detector ratios given above were recalculated. The perturbed detector ratio, *D*, resulting from the uniform removal of fuel rods is given in Eq. (A.9):

$$D_x^{(k)} = \frac{C_i^{(k)}}{C_j^{(k)}},\ \sigma_{D(x,k)} = \frac{C_i^{(k)}}{C_j^{(k)}}\sqrt{\left(\frac{\sigma_i^{(k)}}{C_i^{(k)}}\right)^2 + \left(\frac{\sigma_j^{(k)}}{C_j^{(k)}}\right)^2} \tag{A.9}$$

where

k = 1, 2, 3, corresponding to the region from which fuel rods were removed from the assembly
x = 1,…, 6, corresponding to the six detector ratios given in Eq. (A.3) – (A.8)

The subscript *i* corresponds to the detector used in the *numerator* of the six ratios where *i = B_4C ^{235}U or Bare ^{235}U fission chambers*. The subscript *j* corresponds to the detector used in the *denominator* of the six ratios where *j = Bare ^{235}U, Gd covered,* or *Cd covered fission chambers*.

To assess the sensitivity of each region in the assembly to the uniform removal of fuel rods, the percent-difference, *P*, between the detector ratio, *R* (*no defects*), and the detector ratio, *D* (*partial defects*), and corresponding standard deviations were calculated for each region (*k*) using the following equations:

$$P_x^{(k)} = \frac{R_x^{(k)} - D_x^{(k)}}{R_x^{(k)}} \times 100$$

$$let\ A = R_x^{(k)} - D_x^{(k)}\ and\ \sigma_A = \sqrt{\left(\sigma_{R(x,k)}\right)^2 + \left(\sigma_{D(x,k)}\right)^2} \tag{A.10}$$

$$\Rightarrow \sigma_{P(x,k)} = \frac{A_x^{(k)}}{R_x^{(k)}}\sqrt{\left(\frac{\sigma_{A(x,k)}}{A_x^{(k)}}\right)^2 + \left(\frac{\sigma_{R(x,k)}}{R_x^{(k)}}\right)^2}$$

Quantitative Studies to Detect Partial Defects in Spent Nuclear Fuel Using the Digital Cerenkov Viewing Device

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Abstract:

The Digital Cerenkov Viewing Device (DCVD) has been successfully used for verification of spent nuclear fuel stored in water ponds. Attention has now been focused on using the DCVD to detect missing and substituted fuel rods in an assembly. To detect missing and substituted fuel rods, the DCVD has to be used in a quantitative manner. This requires characterization of the DCVD for accuracy and precision. In order to obtain an accurate Cerenkov light intensity of a spent fuel assembly, the effect of near neighbours must also be determined. Field tests have been performed to assess the capability of the DCVD to detect missing and substituted fuel rods within an assembly. The preliminary results will be discussed in this paper.

Keywords: partial defect detection; Digital Cerenkov Viewing Device; spent fuel verification

1 Introduction

The Cerenkov image produced by the Digital Cerenkov Viewing Device (DCVD) provides both qualitative and quantitative information. The initial DCVD development studies centred on the qualitative capabilities of the instrument to detect low-burnup and long-cooled spent fuel. Current studies have focussed on the quantitative capabilities of the instrument and its potential to detect missing or substituted fuel rods (partial defects). A partial defect is currently defined by the International Atomic Energy Agency (IAEA) to be 50 percent missing or substituted fuel rods.

Previous field studies have shown that the DCVD is capable of detecting a single missing fuel rod if it is not obscured by the top structure of the fuel assembly[1]. A single substituted irradiated zircaloy rod located on the outside edge of a boiling water reactor fuel assembly can also be detected. These studies were qualitative involving observation of the image of the fuel assembly and visual detection of the anomaly.

The quantitative method for partial defect detection is based on the assumption that the Cerenkov flux of a spent fuel assembly is proportional to its burnup and cooling time. A fuel assembly with substituted rods will have less gamma emitting fission products and thus a correspondingly lower Cerenkov flux. The change in Cerenkov flux from the fuel assembly should be measurable using the DCVD.

Measurements have been made to determine the precision and accuracy of the DCVD. Field studies on both pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel assemblies were carried out to determine the degree of uncertainty caused by the precision of measurement, alignment of the DCVD over a fuel assembly, different fuel design and the effect of near neighbours.

The objective of this study is to determine the ability of the DCVD to detect the difference between a normal spent fuel assembly and a fuel assembly with 50 percent substituted stainless steel fuel rods.

2 Image Evaluation Method

The Cerenkov image of a spent fuel assembly is taken with the DCVD aligned axially with the fuel assembly. The resulting Cerenkov image (Figure 1 centre) shows dark areas for the top structure and fuel rods and brighter areas for the water gaps between the fuel rods.

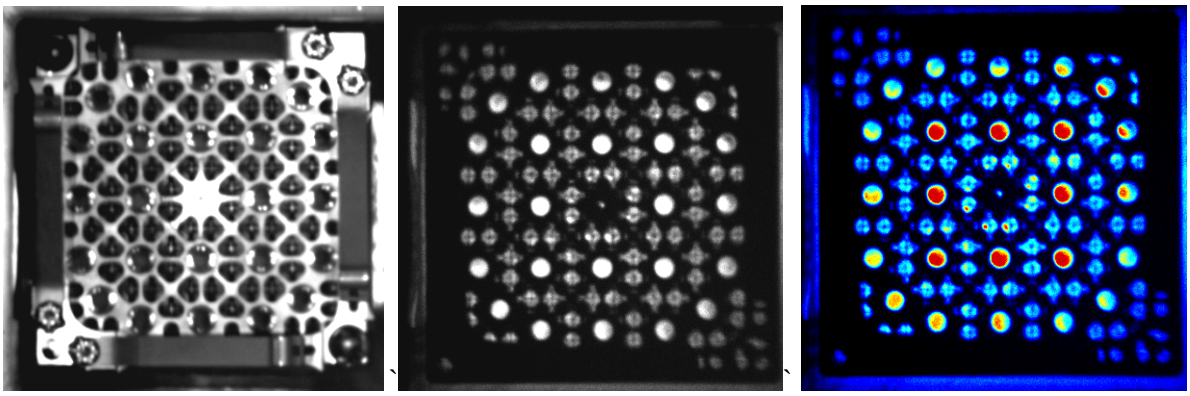

Figure 1: Visible image (left), Cerenkov image (centre), and false coloured Cerenkov image (right) of a PWR 17x17 spent fuel assembly

Previous computer modelling has shown that the bright pixels in the Cerenkov images have contributions from the entire length of the fuel assembly [2]. The intensity of the Cerenkov light in the water gaps is proportional to the gamma radiation intensity generated by the fuel assembly.

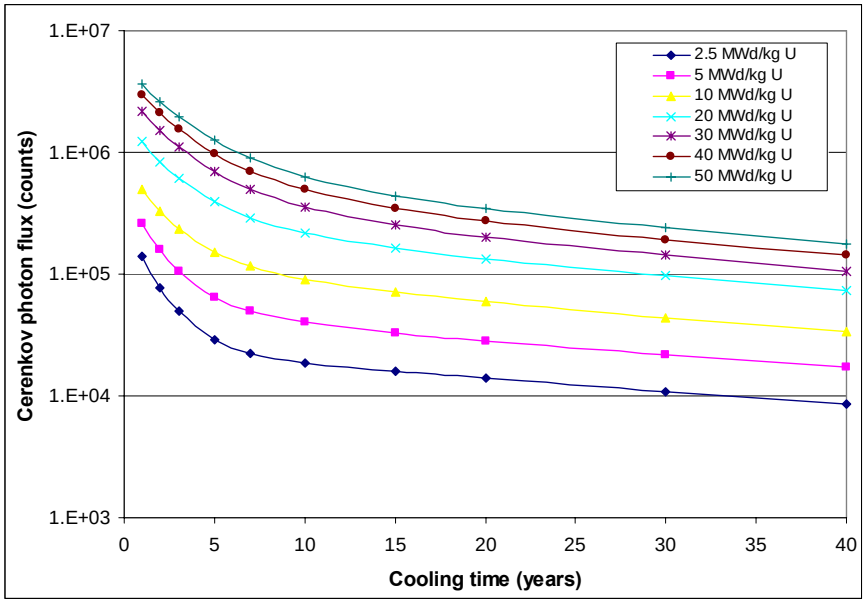

Figure 2: Photon flux from BWR fuel as a function of burnup and cooling time

2.1 Factors influencing the intensity of a Cerenkov image

The number of Cerenkov photons generated by a fuel assembly is a direct result of the assembly's burnup, cooling time and irradiation history. The calculated Cerenkov photon flux is based on theoretical and experimental measurements on BWR fuel by S. Rolandson[3] and L. Ilver[4]. The calculated Cerenkov photon fluxes as a function of cooling time and burnup are shown in Figure 2. Because there are no theoretical calculations for the PWR fuel assemblies, estimated values were interpolated from the BWR curves.

The number of Cerenkov photons generated by a spent fuel assembly is reduced by absorption by the water in the fuel pond before reaching the DCVD detector. The DCVD also filters the Cerenkov light so that only a narrow portion of the Cerenkov spectrum, from 280 nm to 320 nm, reaches the detector.

The Cerenkov light intensity emitted by a fuel assembly is also affected by Cerenkov photons generated by gamma rays from neighbouring fuel assemblies, the alignment of the instrument above the fuel assembly, stray ultraviolet photons produced by ambient lighting and water turbulence.

3 Calculation method for Cerenkov light intensity

The Cerenkov light intensity of a DCVD fuel assembly image is calculated using the following procedure:

- A region of interest (ROI) is selected from the image that includes the fuel assembly in question and as little of the surrounding area as possible.
- A background value, the minimum average intensity of each 3x3 pixel area in the selected ROI, is determined.
- This background value is subtracted from each pixel in the ROI to obtain a corrected intensity value. (This removes the camera background level and the average stray light in the image)
- The corrected intensity values of all pixels in the ROI are then summed to produce the measured Cerenkov light intensity value from the assembly.

The code to implement the intensity calculation is written in LabVIEW V8.6. The code is designed to allow the user to select the ROI of interest to determine the Cerenkov intensity of the fuel. The results are automatically placed in a Microsoft Excel spreadsheet for further analysis.

3.1 Measurement precision

PWR and BWR fuel assemblies were measured to determine the measurement reproducibility of the DCVD. The measurement method involved ten measurements of a single fuel assembly, moving the DCVD away from the assembly and realigning the instrument each time, to simulate a real measurement situation. A total of eight PWR fuel assemblies and nine BWR fuel assemblies were measured in this manner. Typical results are shown in Table 1.

Table 1: Typical precision results from PWR and BWR fuel assemblies

Fuel ID, PWR	**Intensity (counts)**	**Fuel ID, BWR**	**Intensity (counts)**
4D2, 17x17 W	1.36×10^8	18077, SVEA 64	3.61×10^7
43 813 MWd/t U	1.38×10^8	37 877 MWd/t U	3.84×10^7
cooled 7 years	1.38×10^8	cooled 7 years	3.79×10^7
	1.40×10^8		3.72×10^7
	1.39×10^8		3.69×10^7
	1.39×10^8		3.72×10^7
	1.41×10^8		3.81×10^7
	1.41×10^8		3.74×10^7
	1.42×10^8		3.79×10^7
	1.43×10^8		3.75×10^7
Mean	**1.40×10^8**	**Mean**	**3.75×10^7**
Precision (2σ)	**4.09×10^6**	**Precision (2σ)**	**1.33×10^6**
Precision (2σ)	**2.92%**	**Precision (2σ)**	**3.55%**

The PWR fuel assemblies had an average burnup of about 45 000 MWd/t U and the cooling time ranged from 7 years to 17 years. The BWR fuel had an average burnup of about 34 000 MWd/t U and the cooling time ranged from 2 to 18 years. The average precision (2 sigma) for PWR fuel was ±3.2 percent and for BWR fuel ±3.6 percent.

3.2 Instrument alignment

The effect of instrument alignment on the measurement precision was investigated by moving the DCVD from the left to the right side of the fuel assemblies in one-centimetre increments. The results for a typical PWR fuel assembly are shown in Figure 3. The results show that if the DCVD is aligned

within ±5 cm of the alignment point the difference in measured intensity is less than 5 percent of that obtained when the instrument is correctly aligned. The results for a BWR fuel assembly show similar results.

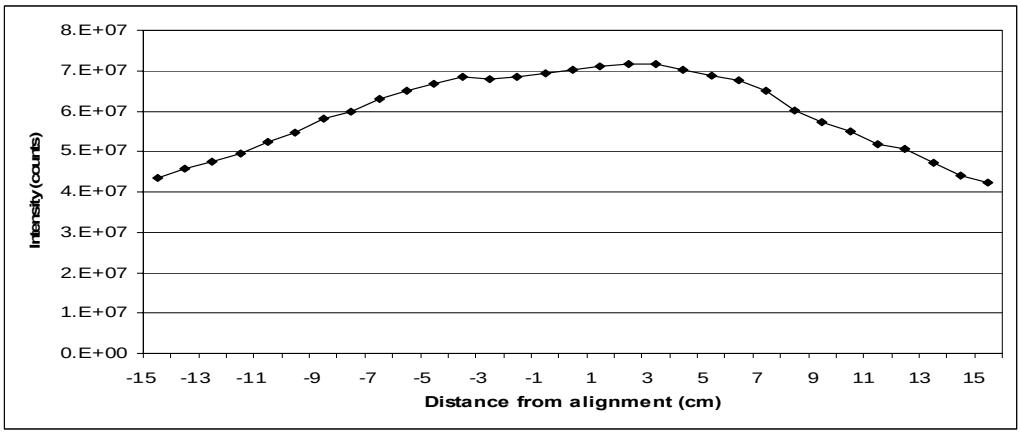

Figure 3: Cerenkov image intensity relative to alignment position

Figure 4 shows the difference between an aligned image and an image taken 5 cm off alignment. The alignment difference would be easy for an operator to detect visually.

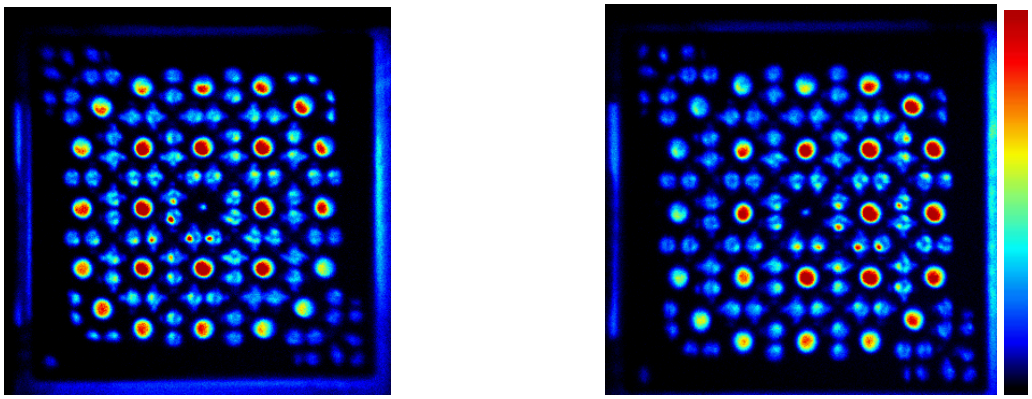

Figure 4: Aligned PWR image (left) and 5 cm off alignment (right)

3.3 Near neighbour effect

The effect of near neighbour fuel assemblies was studied for PWR and BWR fuel assemblies. The experiment to measure the effect of three neighbouring PWR fuel assemblies and four BWR assemblies is shown in Figure 5. The PWR fuel assembly 3X2 (45,341 MWd/t U, cooled 2.3 years) was measured in an isolated location and then with 1, 2 and 3 near neighbours.

The three neighbouring assemblies have almost identical burnup and cooling times. The storage rack pitch in the fuel bay is 35 cm and the width of a PWR fuel assembly is about 21 cm. The storage rack pitch for the BWR fuel pond is 25 cm and the width of the fuel assembly is about 14 cm. The fuel assembly K00019, 41 283 MWd/t U, cooled 2.3 years was measured isolated and with 1 to 4 near neighbours with similar burnup and cooling times.

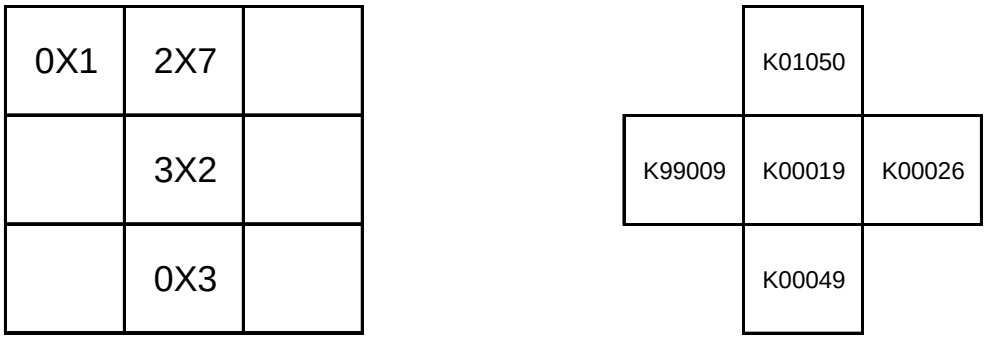

Figure 5: Near-neighbour arrangement for PWR fuel (left) and BWR fuel (right)

The resulting Cerenkov intensities are given in Table 2. The results show a small increase in intensity; however, the differences are within the DCVD measurement precision of ± 5%.

Table 2: Cerenkov intensities with near neighbours

PWR 3X2 assembly (centre)	**Intensity (counts)**
Isolated	$7.82 \times 10^7 \pm 5\%$
One near neighbour	$7.54 \times 10^7 \pm 5\%$
Two near neighbours	$7.74 \times 10^7 \pm 5\%$
Three near neighbours	$7.84 \times 10^7 \pm 5\%$
BWR K00019 centre assembly	**Intensity counts**
Isolated	$2.78 \times 10^7 \pm 5\%$
One near neighbour	$2.80 \times 10^7 \pm 5\%$
Two near neighbours	$2.85 \times 10^7 \pm 5\%$
Three near neighbours	$2.88 \times 10^7 \pm 5\%$
Four near neighbours	$2.93 \times 10^7 \pm 5\%$

For both PWR and BWR spent fuel assemblies with similar burnup and cooling times the effect of near-neighbours is statistically within the measurement error of the instrument.

4 Quantitative measurement of a grid of assemblies

A grid of PWR and BWR fuel assemblies were quantitatively analyzed to determine whether it is possible to detect a 50 percent fuel rod substitution. This degree of substitution should produce a corresponding decrease in Cerenkov light intensity because the substituted rods are not radioactive and therefore do not contribute to the Cerenkov glow emitted by the fuel assembly. To determine whether it is possible to detect 50 percent fuel rod substitution, it is necessary to compare the measured results with theoretical calculated results (Figure 2). If there is a 50 percent fuel rod substitution in a fuel assembly, the measured Cerenkov light intensity should be 50 percent lower than the theoretical calculated result. Modelling work is currently underway to confirm this assumption

4.1 Cerenkov light intensities of PWR spent fuel

Sixteen fuel assemblies in a 4x6 storage grid were imaged. Burnups ranged from 15 000 to 48 000 MWd/t U and the cooling time varied from 0.2 to 14.3 years. Three of the assemblies were not used because they had a cooling time of only two months and there is no modelling data for this time period. Another was a dummy assembly and was not used. The data from Figure 2 was used to interpolate the theoretical (expected) Cerenkov intensities for each of the PWR fuel assemblies. The results are summarized in Table 3.

Table 3: PWR spent fuel intensities and normalized data

PWR assembly information					2Y2, reference	
Fuel ID	Cooling time (years)	Burnup (MWd/ kg U)	Expected Intensity (counts)	Measured intensity (counts ±5%)	Expected difference	Measured difference
2Y2	1.3	46	3.38×10^{6}	8.14×10^{7}	0%	0%
1X2	2.3	46	2.24×10^{6}	6.10×10^{7}	34%	25%
1V4	3.3	45	1.68×10^{6}	3.73×10^{7}	50%	54%
1X3	3.3	44	1.64×10^{6}	4.25×10^{7}	52%	48%
1W2	3.3	47	1.75×10^{6}	4.07×10^{7}	48%	50%
1W7	3.3	48	1.79×10^{6}	4.04×10^{7}	47%	50%
03V	3.3	45	1.69×10^{6}	3.65×10^{7}	50%	55%
0V7	3.3	47	1.75×10^{6}	4.06×10^{7}	48%	50%
02V	3.3	45	1.68×10^{6}	3.80×10^{7}	50%	53%
1X6	3.3	44	1.64×10^{6}	4.58×10^{7}	52%	44%
3V6	4.3	45	1.35×10^{6}	2.49×10^{7}	60%	69%
5A0	14.3	23	1.98×10^{5}	5.48×10^{6}	94%	93%

Fuel assembly 2Y2 was arbitrarily used as a reference assembly to normalize the expected (calculated) and measured Cerenkov light intensities for all of the fuel assemblies (Table 3). The expected and measured differences show good agreement. This data indicates that there are no significant partial defects in this grid of fuel assemblies. Figure 6 shows the relationship of expected and measured intensities relative to assembly 2Y2.

Fuel assembly 2Y2 has an expected intensity of 3.38×10^{6} counts (no partial defect) and if we assume that this assembly has a 50 percent partial defect, this assembly should then produce a 50 percent lower light output. In the list of fuel assemblies, IV4 has an expected intensity of 1.68×10^{6} counts which is about half of the expected light intensity of 2Y2. The measured light intensity is 54 percent of 2Y2 which is in excellent agreement with the expected value of 50 percent. These results indicate that that the DCVD should be able to detect a 50 percent decrease in light intensity and therefore a 50 percent partial defect.

If we now assume that 2Y2 has a 30% partial defect, this assembly should produce 30 percent lower light intensity. In the list of fuel assemblies 1X2 has an expected intensity of 2.24×10^{6} counts which is 34 percent lower in light intensity and the measured decrease in light intensity is 25 percent which is in fair agreement. This data indicates that it may be even possible to detect a 30 percent partial defect.

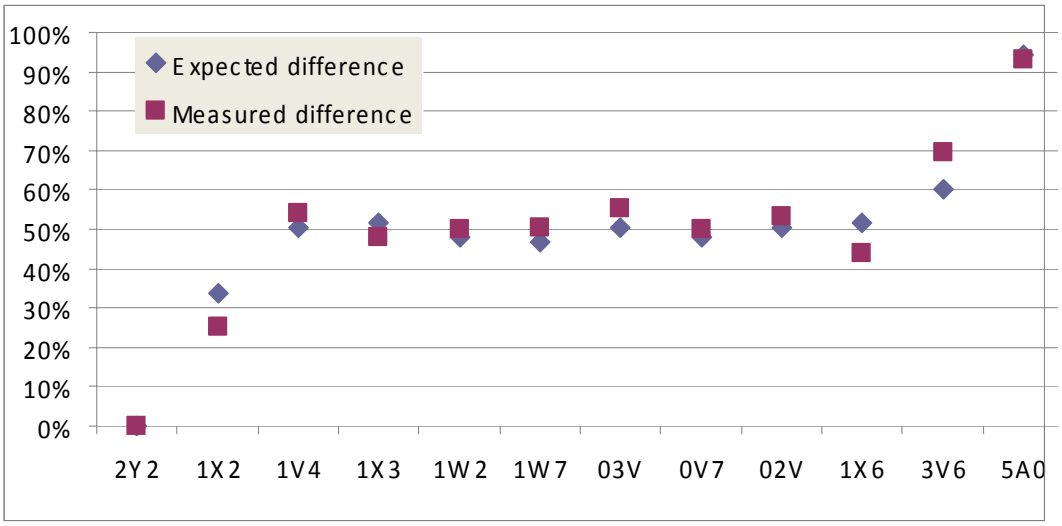

Figure 6: Expected and measured intensities relative to PWR assembly 2Y2

4.2 Cerenkov light intensities of BWR spent fuel

For the BWR study a 5x8 grid of BWR fuel assemblies was selected. There were 28 fuel assemblies with burnup ranging from 45 000 to 17 000 MWd/t U and cooling time from 0.3 to 28 years. There were four empty sites and eight sites containing fuel channel boxes, which were treated as empty sites.

This exercise is made more complicated because there were four different designs of fuel assemblies. The measured intensity of Atrium 10 fuel was found to be 25 percent higher than the measured intensity for SVEA 96S fuel for assemblies with identical burnup, cooling time and irradiation history. SVEA 96S and SVEA 100 fuel assemblies are similar in design. The data analysis was confined to SVEA fuel assemblies and the results are summarized in Table 4. The expected and measured differences show relatively good agreement. This data indicates that there are no significant partial defects in this group of fuel assemblies. Figure 7 shows the relationship of expected and measured intensities relative to assembly 23829.

Fuel assembly 23829 has a calculated intensity of 1.68 $x10^6$ counts. The other fuel assemblies in Table 4 have about 50 percent lower calculated intensity. These assemblies would be representative of a 50 percent partial defect (50 percent decrease in light intensity). The measured light intensities are around 50 percent lower in light intensity. These results indicate that that the DCVD should be able to detect a 50 percent decrease in light intensity and therefore a 50 percent partial defect.

Table 4: BWR SVEA spent fuel intensities and normalized data

BWR assembly information					23829, reference	
Fuel ID	Cooling time (years)	Burnup (MWd/kg U)	Calculated intensity	Measured intensity	Expected difference	Measured difference
23829	3.2	46	1.68 $x10^6$	2.38 $x10^7$	0	0
21352	6.3	43	8.65 $x10^6$	9.73 $x10^6$	59%	49%
21351	6.3	43	8.65 $x10^5$	8.60 $x10^6$	64%	49%
21344	6.3	44	8.87 $x10^5$	1.07 $x10^7$	55%	47%
21340	6.3	44	8.87 $x10^5$	9.76 $x10^6$	59%	47%
21354	6.3	44	8.87 $x10^5$	8.12 $x10^6$	66%	47%
21335	6.3	45	9.25 $x10^5$	1.11 $x10^6$	53%	45%

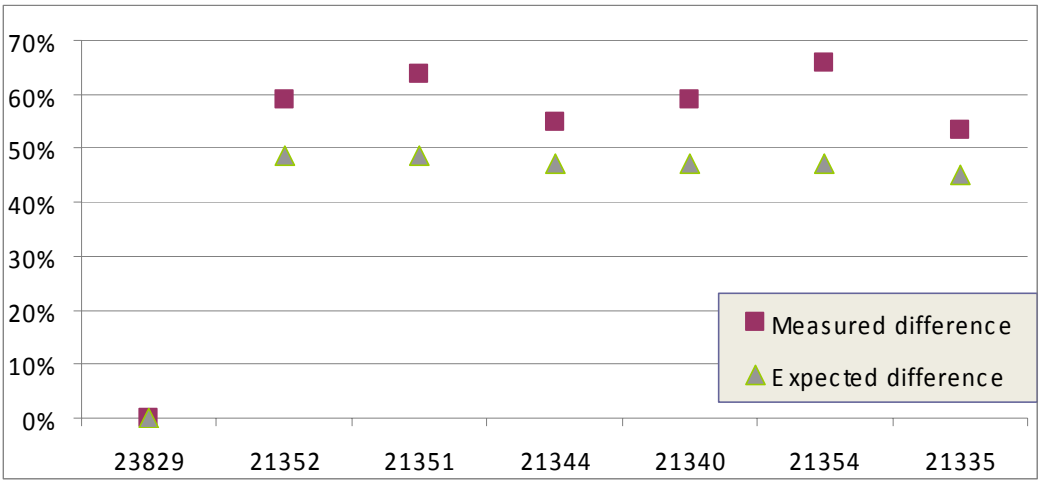

Figure 7: Expected and measured intensities relative to BWR assembly 23829

5 Conclusions

This study indicates that it should be possible to detect a partial defect of 50 percent substituted fuel rods in a spent fuel assembly. The PWR data shows that it may be even possible to detect a 30 percent partial defect in PWR spent fuel.

Modelling efforts are now underway to support the assumption that a 50 percent decrease in fuel pins in an assembly will result in a 50 percent decrease in measured Cerenkov intensity.

Further studies are also required to characterize the Cerenkov light intensity of fuel assemblies with different designs, particularly for BWR fuels. This is the first attempt to quantify the Cerenkov light intensities of BWR fuel assemblies and further studies are required to confirm or improve on the current results.

6 Acknowledgements

The authors are indebted to the Swedish Nuclear Fuel and Waste Management Company, owners of the Swedish Central Interim Storage for Spent Fuel (CLAB) and The Ringhals Group, owner of the Ringhals nuclear power station for making this study possible. The cooperation of their staff for the planning, field measurement and documentation of the results is much appreciated.

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Spent Fuel Temperature and Age Determination from the Analysis of Uranium and Plutonium Isotopics

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Abstract:

The capability to determine the age (time since irradiation) of spent fuel can be useful for verification and safeguards. While the age of spent fuel can be determined based on measurements of short-lived fission products, these measurements are not routinely done nor generally reported. As an alternative, age can also be determined if the uranium (U) and plutonium (Pu) isotopic values are available. Uranium isotopics are not strongly affected by fuel temperature, and burnup is determined from the ^{235}U and ^{236}U isotopic values. Age is calculated after estimating the ^{241}Pu at the end of irradiation while accounting for the fuel temperature, which is determined from ^{239}Pu or ^{240}Pu. Burnup and age determinations are calibrated to reactor models that provide uranium and plutonium isotopics over the range of fuel irradiation. The reactor model must contain sufficient fidelity on details of the reactor type, fuel burnup, irradiation history, initial fuel enrichment and fuel temperature to obtain accurate isotopic calculations. If the latter four are unknown, they can be derived from the uranium and plutonium isotopics. Fuel temperature has a significant affect on the production of plutonium isotopics; therefore, one group cross section reactor models, such as ORIGEN, cannot be used for these calculations. Multi-group cross section set codes, such as Oak Ridge National Laboratory's TRITON code, must be used.

Keywords: forensics, temperature, age, spent fuel, burnup, fuel temperature

1. Introduction

Verification of declared nuclear material is an important and challenging task. Several techniques exist to find the age or cooling time of spent fuel. These techniques are based on measurements of either fission products or other actinides besides uranium and plutonium isotopics. Unfortunately, frequently only the plutonium and uranium isotopics are measured. We have developed a method to determine the age of a spent fuel sample from uranium and plutonium isotopics along with the integral temperature experienced by the sample during irradiation.

2. Preconditions

In order to find the age and fuel temperature from uranium and plutonium isotopics, specific reactor modeling tools and knowledge of the reactor and initial fuel are needed.

2.1. Reactor Modeling Software

Fuel temperature has a significant impact on the production of plutonium isotopics. Not all reactor modeling codes (e.g. ORIGEN) can adjust for differences in fuel temperature. Therefore, only reactor modeling codes that can account for fuel temperature, such as reactor codes that link monte-carlo and depletion methods, are suitable. Two such examples are: Triton, which is contained in Oak Ridge National Laboratory's (ORNL) SCALE package, and Los Alamos National Laboratory's (LANL) latest MCNPX package, which includes the CINDER depletion code.

2.2. Reactor Model Parameters

It is well understood that different reactor types produce differing uranium and plutonium isotopics at the same burnup. At a given burnup, the generated uranium and plutonium isotopics within the fuel are largely determined by the moderator and initial fuel isotopics. Small variations in the exact fuel spatial configuration, fuel diameter and cladding thickness have an important but smaller affect on the uranium and plutonium isotopics. As such, our method doesn't require an exact reactor model that matches the reactor source, but it does require the reactor model to be in the same reactor class (e.g. CANDU, PWR, MAGNOX, RBMK).

2.3. Initial Fuel Isotopics

The initial fuel enrichment has a significant impact on the uranium and plutonium isotopics generated. When the initial fuel enrichment is unknown, there are methods to determine it[1]. However, these are somewhat complicated and are not discussed in this paper. In some cases, the initial fuel isotopics are known, such as in a natural uranium fueled reactor. This may pose a problem when the sample of interest comes from a reactor that uses low-enriched uranium (LEU) fuel, due to the fact that some reactors (e.g. PWRs) use multiple enrichment levels within the same core load and vary the enrichment with time.

Another potentially important parameter is the amount of initial ^{236}U in the fuel. Our method relies on the ^{236}U isotopic abundance of the spent fuel to determine burnup. LEU fuel that originates from the U.S. contains some initial ^{236}U. We found that at the higher burnup range, above 10 GWd/MT, the precise fresh fuel ^{236}U content does not have a significant affect on the burnup determination using the ^{236}U content of the spent fuel. Most reactors that use LEU fuel tend to have burnup ranges much higher than 10 GWd/MT. If a LEU spent fuel sample has an unknown fabrication origin and low burnup, then larger uncertainties for age and fuel temperature determinations will exist.

2.4. Uncertainties on Isotopic Measurements

Various measurement techniques will produce isotopic values with differing uncertainties. Our method requires an uncertainty of approximately 5% or less. With very low burnup samples, the small percentage of plutonium within the fuel may prove to be too difficult to accurately measure the ^{241}Pu concentration with an uncertainty less than 5%. Samples with a burnup greater than 300 MWd/MT are preferable.

3. Temperature and Age Determination

To find the temperature and age (time since the end of irradiation), the following procedure is used.. First, the burnup is determined by using ^{236}U. Next, the fuel temperature is calculated by using ^{239}Pu and ^{240}Pu. Then, the ^{241}Pu at the end irradiation is estimated by using the previously determined burnup and fuel temperature. Finally, the age is calculated by using the estimated ^{241}Pu isotopic abundance at the end of irradiation and the measured ^{241}Pu isotopic abundance.

3.1. Burnup

The burnup is determined using an isotope that is temperature independent. Both ^{236}U and ^{235}U meet this requirement and can be used. Determination of burnup by ^{235}U is more sensitive to small uncertainties in the measurement, so determining the burnup by ^{236}U is preferable. As stated above, we are assuming knowledge of the reactor type and initial fuel isotopics to create a Triton reactor pin model. Calculations are done to ensure the depletion steps are no larger than 50-100 MWd/MT. A plot of the burnup as a function of the production of ^{236}U is given in Figure 1. A curve fit of calculated ^{236}U abundance can then be used to determine the burnup for the measured ^{236}U. A similar process can be used with ^{235}U if a comparison is desired.

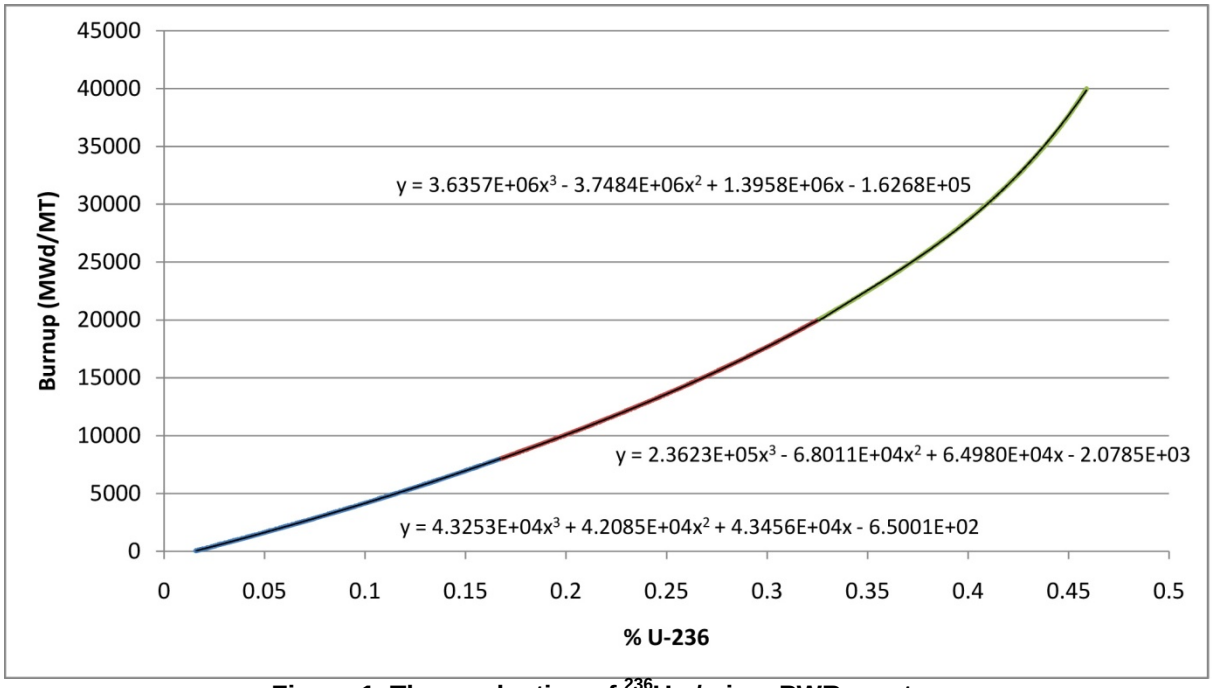

Figure 1. The production of ^{236}U a/o in a PWR reactor.

3.2. Fuel Temperature

To estimate the fuel temperature, the ^{239}Pu and ^{240}Pu isotopics are calculated at various temperature intervals by using the Triton reactor model. Figure 2 shows how ^{239}Pu varies with temperature in a PWR reactor. At 30,000 MWd/MT, the difference in ^{239}Pu abundance at 150° C and 650° C is an absolute difference of 7%. However, by using the burnup found from ^{236}U, the temperature can be estimated by interpolating between the varying temperature curves. Two independent temperature estimates can be found by applying this method to ^{239}Pu and ^{240}Pu separately and then compared. If temperature estimates differs by more than 50°C, then the isotopic values may be of poor quality.

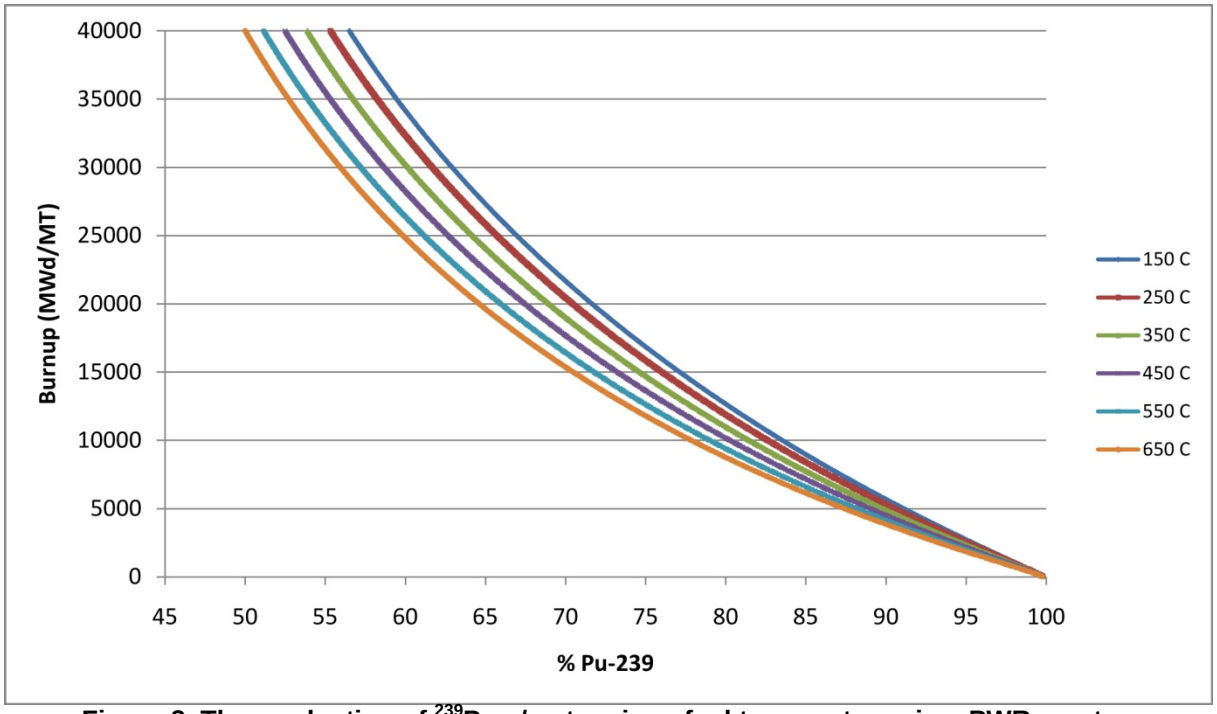

Figure 2. The production of ^{239}Pu a/o at various fuel temperatures in a PWR reactor.

3.3. Age

Once the burnup and fuel temperature estimates have been determined, it is then possible to predict the ^{241}Pu abundance at the end of irradiation from reactor modelling as shown by Figure 3. To avoid heavy computational time for each sample, a series of reactor modelling runs may be computed at varying temperatures, similar to ^{239}Pu and ^{240}Pu estimates. Then, the $^{241}Pu_o$ may be estimated by interpolating between the temperature-dependent burnup curves at the correct burnup. If the reactor is operated at a significantly higher or lower specific power than expected, then this procedure will produce a higher uncertainty in the ^{241}Pu estimate.

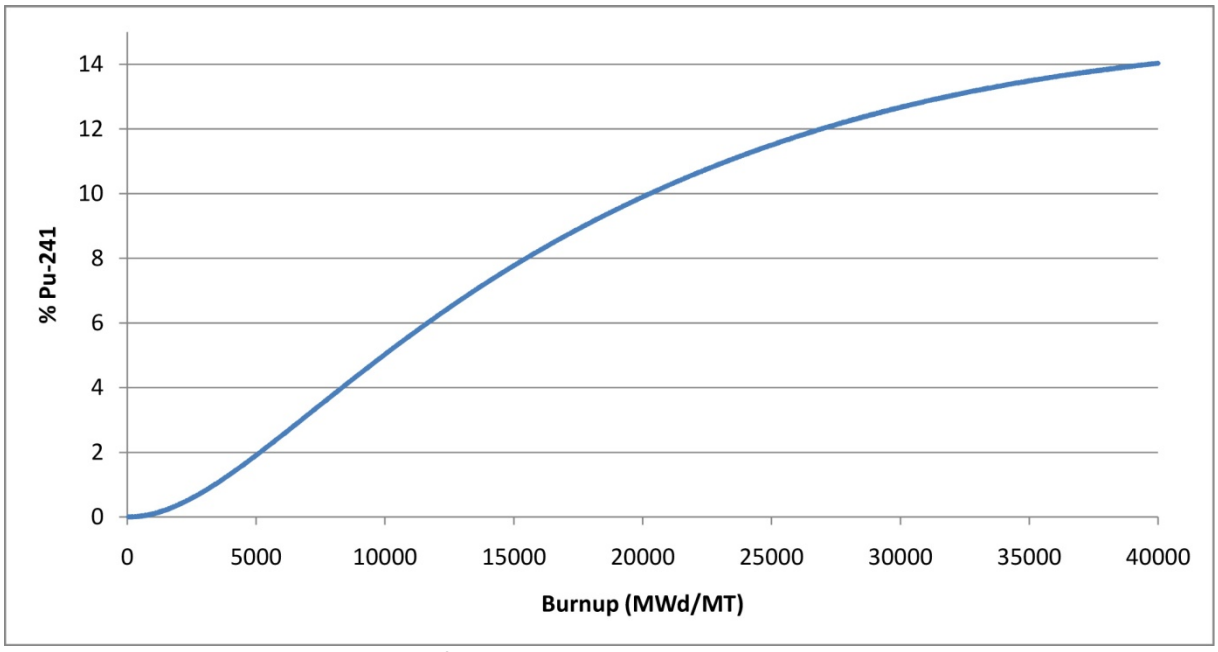

Figure 3. The production of ^{241}Pu in a PWR reactor at a fuel temperature of 350° C.

The age or cooling time is found by using the standard decay equation and solving for time.

$$T = -\frac{T^{1}/_{2}}{\ln(2)}\ln\left(\frac{241Puo}{241Pu(T)}\right),$$

Where T is the age, T1/2 is the half life of ^{241}Pu, ^{241}Puo is the atom percent of ^{241}Pu at the end of irradiation and ^{241}Pu(T) is the measured ^{241}Pu atom percent.

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SESSION 14

REMOTE TECHNIQUES FOR TREATY VERIFICATION

High-Resolution Radar Imagery for Safeguards Applications

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Abstract:

In support of the International Atomic Energy Agency (IAEA) the Canadian Safeguards Support Program (CSSP) is continuing to investigate techniques to exploit high-resolution synthetic aperture radar (SAR) imagery for nuclear safeguards applications. The paper will include a review of the commercially available high-resolution SAR sensors and case studies to demonstrate potential applications for safeguards with emphasis on object identification and change detection capabilities. Coherent change detection (CCD) techniques have been used in a previous study with 8 metre data (RADARSAT-1). This study will build upon the previous CCD results to illustrate the benefits of high-resolution 1 metre SAR imagery (e.g. TerraSAR-X). In addition, subsidence mapping will also be performed on the high-resolution SAR data to show how this additional information can be obtained and how they can be applied to nuclear safeguards. The results of the study will be evaluated against other information such as available ground truth and optical imagery.

Keywords: SAR satellite imagery, safeguards application, coherent change detection, CCD, synthetic aperture radar

1. Introduction

The Canadian Safeguards Support Program (CSSP) has been investigating synthetic aperture radar (SAR) imagery for nuclear safeguards applications in support of the International Atomic Energy Agency (IAEA). The commercial availability of high-resolution (1 metre) SAR data in recent years has enhanced the usefulness of SAR imagery for infrastructure analysis and surface change detection.

In a SAR imaging system, the sensor (either space-borne or airborne) transmits radar waves towards a region on the ground. The radar waves will interact with the objects and features on the surface and depending on the geometry and texture, part of the radar energy will scatter away, while part of the radar energy will scatter back towards the sensor [1]. The sensor receives the backscattered waves which are then processed into a SAR image.

One of the more important characteristics of SAR for the monitoring of sensitive sites is that SAR is an active imaging system and emits its own radar waves, unlike optical systems which rely on the sun for illumination. This allows SAR sensors to produce images at night, in conditions of cloud cover, and during weather conditions that would obscure optical systems. Furthermore, because SAR records the magnitude and phase of the backscattered waves, additional interferometric imaging products can be generated from repeat-pass SAR imagery. By post-processing the SAR data it is possible to detect subtle surface disturbances, measure sub-centimetre scale ground deformation, or produce detailed elevation models.

2. Commercially Available Sensors and Characteristics

Since 2007 a new generation of commercial SAR satellites have been launched. These new SAR sensors have higher resolution and improved orbit information and control, which is important for

generating interferometric image products. The following is a summary of the current high-resolution SAR sensors that are commercially available.

RADARSAT-2 is a fully polarimetric C-band SAR satellite that is operated by MacDonald, Dettwiler and Associates Ltd (MDA) and was launched in December 2007. RADARSAT-2 has a ground resolution up to 3x1metres (range x azimuth) and when operating in quad polarized mode has a best resolution of 8 metres. The repeat pass time of this satellite is 24 days.

COSMO-SkyMed is a constellation of X-band SAR satellites that are operated by the Italian Space Agency (ASI). The first three satellites were launched in June 2007, December 2007, and October 2008, with the fourth planned to launch in 2010. COSMO-SkyMed has ground resolution up to 1 metre, and the repeat pass time between satellites is 8 days.

TerraSAR-X is an X-band SAR sensor that is operated by the German Aerospace Center (DLR) and was launched in June 2007. It has a spotlight beam mode with a resolution up to 1 metre and has a repeat pass time of 11 days. Another SAR satellite very similar to TerraSAR-X called TanDEM-X (TerraSAR-X add-on for Digital Elevation Measurement) is expected to launch in late 2009. These satellites will operate in a closely controlled formation approximately 250 to 500 metres apart and are ideal for generating detailed digital elevation models (DEMs).

3. Case Studies

This section presents three case studies that illustrate the application of high-resolution SAR imagery for nuclear safeguards applications.

3.1 Infrastructure Analysis using High-resolution SAR Imagery

Figure 1 shows a WorldView 0.5 metre optical image acquired on Aug. 18, 2008 and 1 metre TerraSAR-X image acquired on Feb. 14, 2009 showing part of the Olympic Dam uranium mine in Australia. Olympic Dam is an underground mine situated over the world's largest known single deposit of uranium [2]. Above ground, the site contains large tailings and evaporation ponds, a quarry, and a processing plant.

Figure 1: WorldView 0.5 metre optical image acquired on Aug. 18, 2008 (left) and TerraSAR-X 1 metre image acquired on Feb. 14, 2009 (right) showing the Olympic Dam mine in Australia.

Previous analysis of this site in early 2007 included 8 metre SAR imagery from RADARSAT-1, which was the highest resolution commercially available at the time. At 8 metre resolution, infrastructure analysis at this mine was limited as only large structures and features were identifiable. The recent commercial availability of high-resolution SAR has significantly increased the usefulness of SAR imagery for detailed infrastructure analysis. Figure 2 shows part of the processing plant at the mine, and the pipes (bright lines, A), circular structures (dark lines, B) and roads (C) are easily visible in the 1 metre TerraSAR-X imagery.

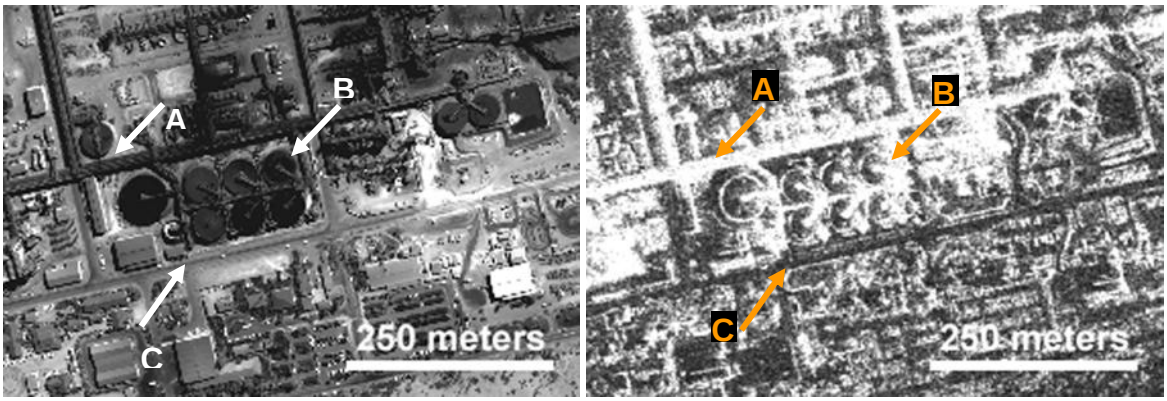

Figure 2: Processing infrastructure shown in WorldView (left) and TerraSAR-X (right).

Figure 3 shows a portion of the quarry at the mine using both optical WorldView and TerraSAR-X imagery. A fence is visible to the right of the quarry and the radar shadows in the right image of Figure 3 allow for easy counting of the three tiers of the quarry. These shadows occur because the radar is acquiring the image at an angle, approximately 53 degrees from normal to the ground for this image. The side of the quarry closest to the satellite blocks the radar signal and the result in the image is a black radar shadow. This is similar to the shadow in the optical image of Figure 3, except that in this case the side of the quarry blocks the sunlight.

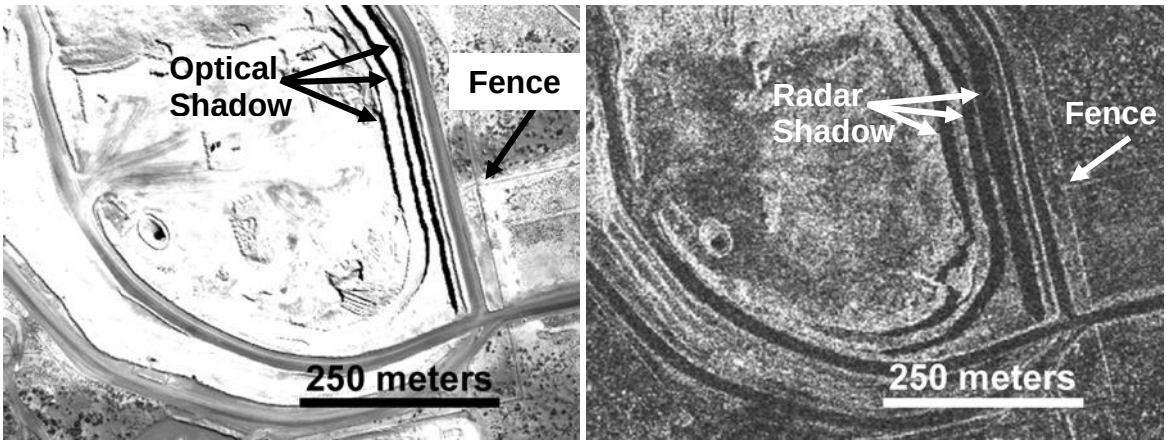

Figure 3: WorldView (left) and TerraSAR-X (right) of the quarry at Olympic Dam mine.

3.2: Detection of Mining Activity

The Beverley Mine in Australia is an in-situ leaching uranium mine operated by Heathgate Resources. The in-situ leaching mining method involves injecting the ore body with a solution that dissolves the ore. This solution circulates throughout the uranium ore body and is extracted at various locations for further processing. In 2002, in-situ leaching accounted for 18.3% of the worldwide production of uranium [3]. As a consequence of this method, there are no large open pits or tailings piles that would indicate if the mine is currently active. One indicator of an operational in-situ leaching mine is the pattern the injection and extraction wells make as they progress along the ore body.

The detection and monitoring of activity at mines is useful for providing information to inspectors, for the verification of declared activities, and for the detection of potential clandestine operations. The Beverley Mine was chosen as a case study because it is in a dry and arid environment, which is ideal for analysis using coherent change detection (CCD). CCD measures the correlation of the backscattered radar waves between two repeat pass SAR images and displays the results in a greyscale image. Black areas have low coherence and represent surface changes while white areas have high coherence and represent areas that have remained the same. CCD uses both the magnitude and phase of the radar image and as a result is able to detect changes that are not visible in traditional SAR images. Low coherence is also generated from the growth and movement of vegetation, erosion and weathering, and from changes in moisture, so interpretation is necessary to distinguish natural changes from those due to human activity.

A previous CCD study of the Beverley Mine was conducted using 8 metre RADARSAT-1 imagery [4]. This study showed that surface disturbances were detected at the mine, although the lower resolution coherence map was not very detailed. The current study was conducted with 1 metre TerraSAR-X Spotlight mode imagery collected on May 16, 2008, May 27, 2008 and October 17, 2008. A section of the Beverley Mine is shown in Figure 4. The left image is a 0.5 m WorldView optical image acquired on March 12, 2008, and the right is a 1 metre TerraSAR-X image acquired on May 16, 2008. The area outlined by the square is shown in more detail in Figure 5 and Figure 6.

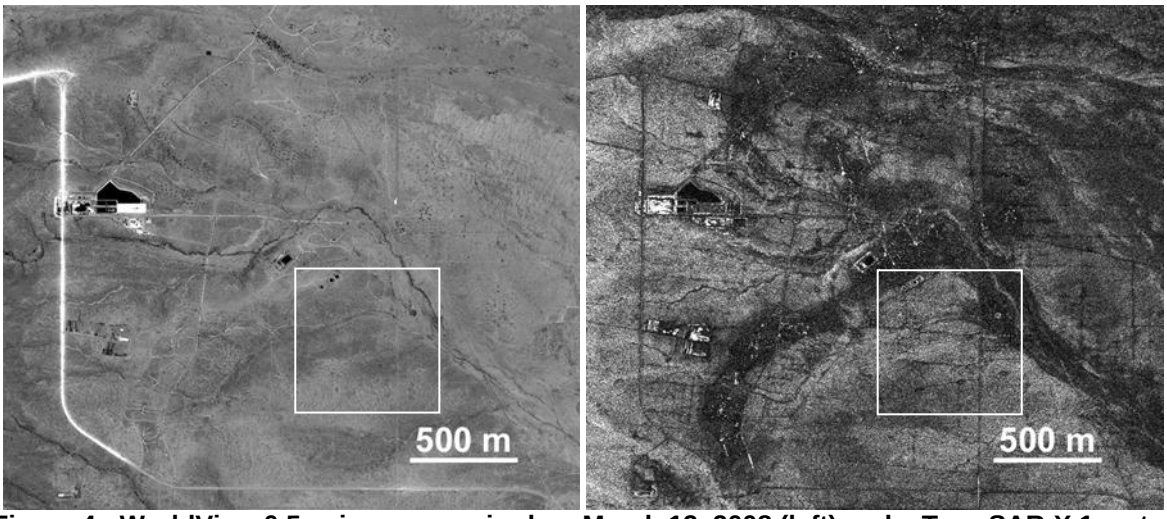

Figure 4: WorldView 0.5 m image acquired on March 12, 2008 (left) and a TerraSAR-X 1 metre resolution image acquired on May 16, 2008 (right) showing part of the Beverley Mine in Australia. The white square shows the location of the CCD analysis in Figure 5 and Figure 6.

Figure 5 shows the changes found using CCD for the TerraSAR-X scenes acquired on May 16, 2008 and May 27, 2008, (left and center images) and represents the surface disturbances during these 11 days (right image). The colour bar on the CCD image shows the value of coherence. Region 1 shows an area where the changes are obvious from the two SAR images, as a grid of points appear in the centre image. These changes are reflected in the CCD image as the large dark area. Region 2 shows an area where there is no observable difference between the two SAR images; however, the CCD image shows dark circles in a grid pattern that indicates surface disturbances.

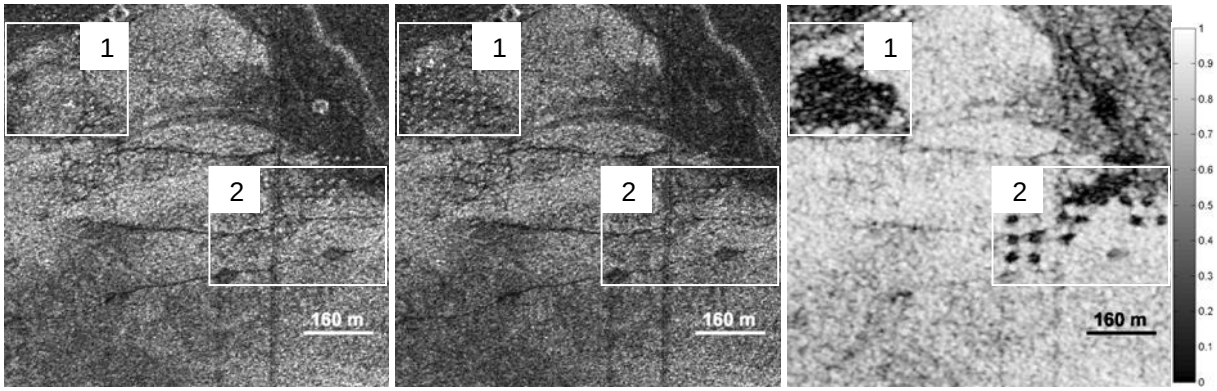

Figure 5: TerraSAR-X images acquired on May 16, 2008 and May 27, 2008 (left, center) of the Beverley Mine. The right image is the CCD map generated from the two TerraSAR-X images.

Figure 6 shows the CCD results of the 154 day interval image pair acquired on May 16, 2008 and Oct 17, 2008 (left and center images). Region 1 shows an area of large changes that is visible in the SAR images, and is also detected in the CCD map (right image of Figure 6). Again, region 2 shows an area where there are no obvious changes in the two SAR images, but the CCD shows more disturbances in the same grid pattern. This pattern of activity could result from the addition of injection, extraction, or exploratory wells involved in the in-situ leaching process.

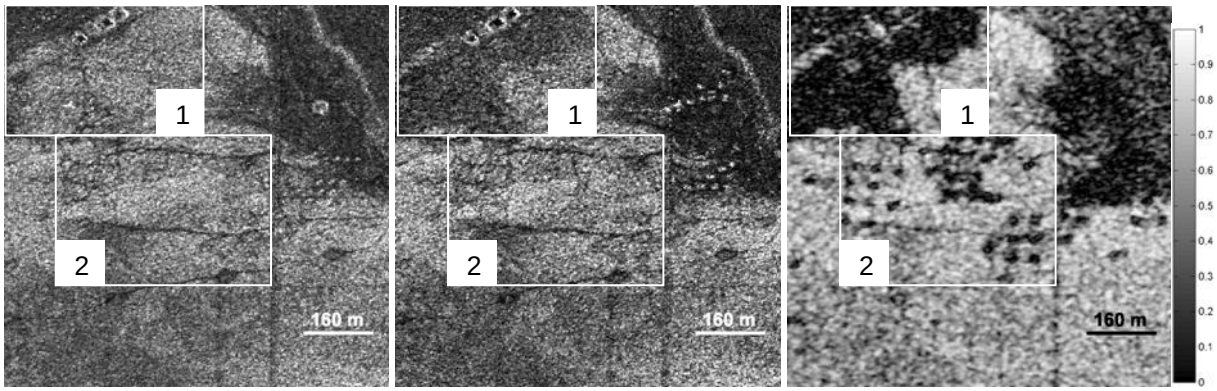

Figure 6: TerraSAR-X images acquired on May 16, 2008 and Oct 17, 2008 (left, center) of the Beverley Mine. The right image is the CCD map generated from the two TerraSAR-X images.

3.3: Detecting Evidence of Mining Activity Based on Ground Movement

Mine subsidence is a well known phenomenon that involves the settling of the ground surface as a result of changes made underground during the extraction of the ore. Typically, subsidence is found at abandoned underground mines where the deformation is due to the collapse of underground tunnels and the settling of displaced material and is usually a gradual change that occurs over time [5].

Subsidence mapping has applications for nuclear safeguards and the verification of declared activities. Ground subsidence in the vicinity of a facility could be an indication of current underground activity or that this area was mined in the recent past. Unexpected changes in the rates of subsidence at former mines or mines that have been declared closed may indicate that there is new activity at the site and that further investigation is required.

SAR is able to measure sub-centimetre deformations in the Earth's surface through differential interferometry (DInSAR) which uses the phase difference between repeat pass SAR imagery to measure the changes in ground height. The benefit of using SAR for measuring subsidence as opposed to conventional techniques such as GPS is that DInSAR can remotely measure small ground deformations over a large area. Although mine subsidence is typically associated with abandoned underground mines, subsidence can also occur at in-situ leaching sites that have already been mined for ore [6]. It is also possible that small deformations of the ground around active injection and extraction pumps may occur and could be observable using DInSAR techniques and the commercially available high-resolution SAR sensors.

Figure 7 shows a subsidence map generated from two TerraSAR-X spotlight mode images acquired on May 16, 2008 and Oct. 17, 2008 of the Beverley Mine. The subsidence map was overlaid on the SAR image and focused on the ore bodies. The yellow areas represent an upward deformation of the ground during the 14 week repeat pass of about 6 to 11 mm, and the blue areas indicate a subsidence rate of approximately 6 to 14 mm. One explanation of these initial findings is that the area experiencing subsidence is due to the extraction of the mining solution, and that the area showing uplift is an area where the mining solution is being injected. The new activity identified in the coherence map of the previous section agrees with this interpretation, as the activity could be from the installation of injection wells. As with all DInSAR analysis, additional SAR imagery and ground truth would be necessary to confirm these results and eliminate uncertainties such as atmospheric effects.

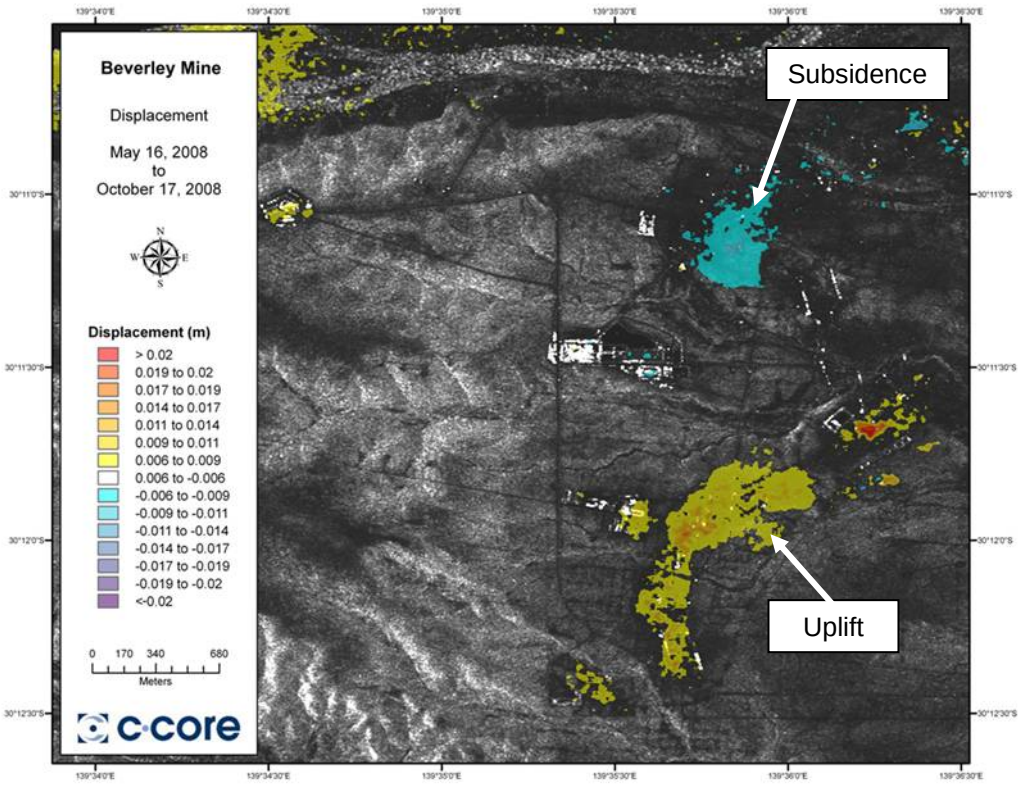

Figure 7: Subsidence map of the Beverley Mine in Australia, generated from TerraSAR-X spotlight imagery.

4. Conclusions

The commercially available high-resolution SAR satellites have a role to play in nuclear safeguards monitoring. The ability of SAR to acquire an image regardless of time of day and cloud cover means that high-resolution (up to 1 metre) imagery can be obtained when it is required. Acquiring multiple, repeat-pass SAR images also means that additional information, such as surface disturbances, ground subsidence, and detailed height information can be generated at sites of interest.

The high-resolution SAR imagery has applications for infrastructure analysis that were previously unavailable with lower resolution SAR. Higher levels of detail allow for easier interpretation of buildings and infrastructure in SAR images by analysts and end users. Interferometric techniques at these high-resolutions allow for the detection of smaller disturbances and ground deformation that occur at sites of interest.

In the presented case studies, complementary optical imagery has been used as ground truth to aid in the analysis of the SAR images. Optical imagery has higher resolution and is easier for a user to interpret. However, due to cloud cover, the acquisition of optical imagery cannot be guaranteed as it can with SAR and this means that current optical imagery may not be available when needed. In the absence of other sources of ground truth, older optical imagery is still relevant for aiding in the interpretation of newer SAR imagery.

The COSMO-SkyMed and proposed RADARSAT constellation missions offer a quicker repeat pass time by using multiple SAR sensors equally spaced in orbit. This quicker revisit time is an important feature for monitoring and means that analysis of mining activity can be observed with smaller intervals which leads to a more detailed knowledge of activities over time. The tandem TerraSAR-X mission will see a second, very similar satellite following closely behind in a closely controlled formation. This configuration will allow interferometric image pairs to be acquired simultaneously. Although this will not be useful for activity monitoring with CCD, it is an ideal situation for the generation of high-resolution digital elevation models.

5. Acknowledgements

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High-resolution SAR Satellite Imagery Analysis for Nuclear Safeguards Applications

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Abstract:

For monitoring nuclear sites in cloudy areas, the use of Synthetic Aperture Radar (SAR) imagery shows essential promises. Unlike optical remote sensing instruments, radar sensors operate under almost all weather conditions and independently of the sunlight, i.e. time of the day. Such technical specifications are required both for continuous and for ad-hoc, timed surveillance tasks. With Cosmo-Skymed, TerraSAR-X and Radarsat-2, high-resolution SAR imagery with a spatial resolution up to 1m has recently become available. Besides the improved spatial resolution, the SAR sensors involve also innovative technical capabilities and thus represent an optimal data source for monitoring nuclear sites. So much for the theory. In practice, however, the operational value of SAR sensors still has to be approved.
Our project therefore aims to investigate the potential of high-resolution SAR data for nuclear monitoring as to the extraction of digital elevation models, 2D and 3D scene change detection, and scene classification. This paper focuses on the extraction of DSMs and change detection. Based on two investigation areas, experiments for radargrammetry and non-coherent change detection will be presented and discussed.

Keywords: SAR satellite imagery, SAR interferometry, SAR polarimetry, radargrammetry

1. Introduction

SAR (Synthetic Radar Aperture, Radar = Radio Detection And Ranging) remote sensing is an active imaging method based on the backscattering of microwaves [1]. SAR sensors are equipped with their own electromagnetic source onboard that transmits microwave pulses towards the Earth's surface. Microwaves are able to penetrate clouds, smoke, fog, haze and also rain (dependent on the wavelengths. Thus, in comparison to the passive optical remote sensing sensors that measure the reflected solar radiation in the visible to the middle infrared region of the electromagnetic spectrum, SAR system operate under almost all weather conditions and independently of the sunlight, i.e. time of the day. Nevertheless the atmosphere interacts with microwaves in a complex way [2].

The transmitted energy is partly reflected from the surface towards the sensor and received as a signal. The amount of the returned signal (backscatter) is a function of the target characteristics (surface roughness, dielectric constant, geometric shape and local incidence angle) and the sensor specifications (wavelength/frequency, incidence angle, polarisation and look direction) [2]. Radar data contain complex pixels values, amplitude and phase. The amplitude gives the intensity of the backscatter signal and is used as image brightness for the grey values on SAR images. The phase is an important parameter for interferometric techniques.

Point scatterers, such as metallic objects, only have one dominant signal in an image pixel, but if different scatterering centres are included in one single image pixel, each of them causes individual

backscattered signals interfering positively or negatively while forming the total signal. Due to this, radar images have a characteristic “salt-and-pepper” appearance called speckle that needs to be reduced by appropriate filtering or multi-look processing. [2]

The microwave radiation can be differentiated according to their frequency and polarization. The smaller the frequency, the deeper the penetration into a surface or a volume scattering object. Today, SAR satellite sensors work with frequencies of 8-12.5 GHz (X-band), 4-8 Ghz (C-band) or 1-2 GHz (L-band)[1]. X-band microwaves are backscattered from the upper vegetation layers, such as tree crowns, whereas L-band microwaves is able to penetrate vegetation and are mostly reflected from the terrain surface. Polarization refers to the orientation of the radar beam relative to the Earth’s surface, either Horizontal (H) or Vertical (V). Once a transmitted signal interacts with the target, the polarization may partly change, either from horizontal to vertical or vice versa, dependent on the physical and electrical properties of the target. Radar systems capable of sending and receiving radar waves both horizontally and vertically can produce co-polarized signals (HH, i.e. transmit H - receive H, and VV) and cross-polarized signals (HV and VH). In case of fully polarimetric (so-called quad-pol) datasets four different polarization channels (HH, HV, VV and VH) are acquired per image. The information provided by quad-pol data enhances the possibilities to derive properties of the earth’s surface.

For monitoring nuclear sites in areas with frequent cloudiness and/or near to the polar circle, i.e. with limited sunlight, the use of SAR imagery shows essential promises in view of continuous and ad-hoc, timed surveillance tasks. With Cosmo-Skymed, TerraSAR-X and Radarsat-2, high-resolution SAR imagery with a spatial resolution up to 1m has recently become available. The operational value of SAR sensors within IAEA safeguards still has to be approved. For this reason, our project aims to investigate the potential of high-resolution SAR data for nuclear monitoring as to the following applications:

- Extraction of a Digital Surface Models (DSMs) using radargrammetric and interferometric techniques;
- Detection of small scale surface movements and other 3D changes by applying interferometric techniques;
- change detection by applying coherent and non-coherent techniques;
- land cover/surface classification based on polarimetric techniques;
- fusion of very high resolution optical and high resolution SAR imagery.

This paper focuses on the extraction of DSMs and change detection. The following chapter gives a brief overview on the state-of-the-art of SAR satellite sensors. Chapter three explains the basics of SAR image processing as to DSM extraction and change detection. Based on two investigation areas, experiments for radargrammetry and non-coherent change detection will be presented in Chapter four. The last chapter gives some conclusions and finally discusses the pros and cons of SAR imagery analysis for nuclear safeguards applications.

2. High-resolution SAR sensors

At present, three SAR sensors provide imagery with a spatial resolution from 3 m: The German TerraSAR-X, the Italian Cosmo-Skymed and the Canadian Radarsat-2 (Table 1). TerraSAR-X and Radarsat-2 have been realized as Public Private Partnerships (PPP) between the respective national space agency and industry. Cosmo-Skymed is a national satellite financed by the Italian Space Agency and the Government.

Depending on the acquisition mode, different swath widths and polarization modes are available. For the highest possible spatial resolution (Stripmap Mode), TerraSAR-X offers single polarization (HH or VV) in 1.1 m and dual polarization (HH/VV) in 2.2 m resolution for a scene size of 10 km (cross) by 5 km (along). Cosmo-Skymed provides single polarization (HH or VV) in 1m resolution (scene size 10 by 10 km^2) and dual polarization (HH/VV or HH/HV or VV/VH) in 15m resolution (30 by 30 km^2). Radarsat-2 generates selective single polarization (HH or VV or HV or VH) at 3 m (20 by 20 km^2) and dual polarization (HH/HV or VV/VH) at 8 m resolution (50 by 50 km^2).

[1] The numbers for the frequency bands vary considerably in the literature. The given numbers are listed according to [3].

Fully polarimetric datasets are available from Radarsat-2 at 8 m azimuth (along flight direction) and 12 m range resolution (across the flight direction) and a scene size of 25 by 25 km², while the quad-pol capabilities on TerraSAR are currently being investigated in an experimental mode. As soon as this mode is operationally qualified and the product characteristics have been assessed, also full polarimetric products may become available [4].

Sensor	Company (Country)	Launch	Spatial resolution	Frequency (band)	Revisit time
TerraSAR-X	DLR/Astrium (Germany)	06/2007	1 m	9.65 GHz (X-band)	11 days
Tandem-X	*DLR/Astrium (Germany)*	*2009 ?*	*1 m*	*9.65 GHz (X-band)*	*11 days*
COSMO-Skymed 1-4	ASI (Italy)	1: 06/2007 2: 12/2007 3: 10/2008 *4: 2010 ?*	1 m	9.6 GHz (X-band)	< 1 day (with 4 Sat.)
RADARSAT-2	CSA (Canada)	12/2007	3 m	5.405 GHz (C-band)	?

Table 1: High-resolution SAR imaging sensors (≤ 3m spatial resolution) currently or planned to be in orbit by 2010, ordered by best spatial resolution and launch date.
Standard font type: launched satellite; italic: to be launched.
(Source: Company's websites, http://directory.eoportal.org/, http://www.space-risks.com)

3. Information Extraction from SAR Imagery

3.1 Extraction of a digital surface models using radargrammetric and interferometric techniques

Digital Elevation Models (DEMs) either display the terrain surface including vegetation, buildings and other objects on the surface (Digital Surface Model, DSM) or the pure terrain height (Digital Terrain Model, DTM). Using SAR imagery, DEMs can be generated by exploiting the amplitude (radargrammetry) or the phase of the radar signal (interferometry). Radargrammetry is based on the same approach that is used with optical imagery in photogrammetry [5]: DEMs are extracted from stereo pairs usually acquired from the same side but with different incidence angles. Interferometric SAR (InSAR) analyses the phase difference between two complex SAR images acquired from slightly different points of view. As the phase difference is related to the terrain topography of the area of interest it can be used for DEM extraction. Figure 1 summarizes the processing steps of both techniques, more technical information is also given in [6,7,8,9,10] for radargrammetric and [1,7,8] for interferometric DEM extraction.

The quality of the DEMs depend in either approach on the coherence, i.e. the correlation between the two scenes. Microwave signals backscattered from one point of interest are coherent, if they vibrate in the same phase. This is theoretically the case if the signals are recorded at the same time and under identical sensor parameters. Due to the real recording conditions at different acquisition times and/or different incidence angles, the coherence is usually minimised. As the coherence is affected by several parameters, the decorrelation can be temporal (different backscatter signals due to land cover changes and/or atmospheric conditions at the two acquisition times), geometric (different incidence angles, ascending or descending mode), topographic (surface elevation changes), thermal (system noise), due to changed Doppler centroid and/or the data processing (e.g. inaccurate registration).

Coherence is a measure for the disparity of the stereo pair after precisely matching the two scenes (Figure 1, left) and the accuracy of the interferometry phase (Figure 1, right). The bigger the coherence, the smaller the noise and the better the accuracy of the matched stereo pair and interferogramm respectively.

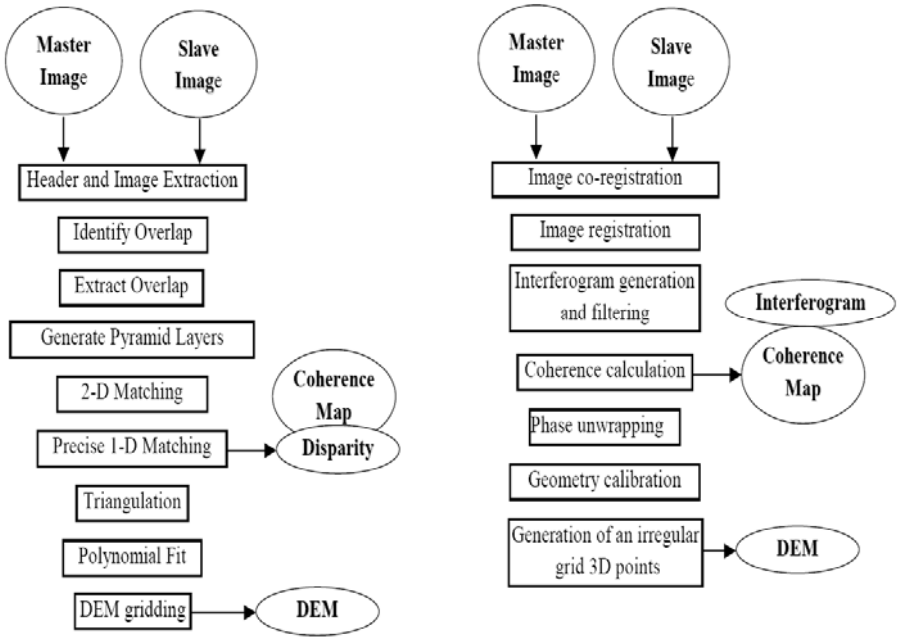


Figure 1: DEM extraction using radargrammetry (left) and interferometry (right) (modified after [6]).

3.2 Change detection by applying non-coherent and coherent techniques

As SAR is a coherent imaging system, two different approaches for surface change detection may be considered: non-coherent and coherent change detection. Non-coherent Change Detection (NCD) identifies changes in the mean backscatter intensity of a repeat pass SAR scene, Coherent Change Detection (CCD) analyses changes in both the amplitude and the phase between two acquisition times. [11] Several techniques exist for either approach.

The easiest approach in NCD is a colour composite of the repeat pass SAR bands, that have been adaptively filtered before to reduce the speckle noise. In a more advanced procedure suggested by [12], the changes are detected based on the filtered intensity or amplitude ratio. Hence, a ratio image is created at first. Since additive noise is easier to filter than multiplicative noise, logarithmic scaling is applied to the ratio image in the second step to make the multiplicative speckle-noise additive. After adaptive filtering, a threshold is estimated in order to differentiate between change and no-change pixels. The method has been successfully implemented in PCI Geomatica and ERDAS Imagine. Other NCD procedures are Constant False Alarm Rate (CFAR) detection, multi-channel segmentation and hybrid methods. [12]

In CCD, the sample coherence of the repeat pass image pair is commonly used to quantify surface changes related to temporal decorrelation. As already stated in Chapter 3.1, temporal decorrelation results from physical surface changes over a specific period of time due to different weather conditions, natural changes of soil moisture, vegetation canopy, surface roughness or topography, or man-made changes. In view of safeguards purposes, human activities such as building and road construction, deforestation, mining etc. are the most interesting changes to be detected. In CCD, the input SAR imagery must be acquired interferometrically, i.e. with the same imaging geometries, and processed interferometrically, including an accurate geometric registration, interferogramm generation and coherence calculation By using the SAR amplitude and the phase, CCD is sensitive to changes in the spatial distribution of scatterers within image pixels and is thus able to find subtle changes that NCD techniques might not detect. [11]

An alternative approach was recently suggested by [11]: In order to discriminate between relevant and irrelevant changes, the change detection procedure can be performed in an hypothesis testing framework. Using e.g. the so-called log-likelihood change statistic, the test determines whether corresponding pixels in an interferometric image pair are realisations of a null (unchanged) hypothesis H_0 or an alternative (changed) hypothesis H_1. Test statistics can be applied to single band and multi band (polarimetric) SAR data [13,14].

4. Experiments

4.1 DSM extraction using radargrammetry

Some experiments on radargrammetry were performed using TerraSAR-X data acquired over the Finnish Olkiluoto peninsula (Figure 2). Surface changes, both two- and three-dimensional, are due to the construction of a third nuclear power reactor and a geological repository for high-level radioactive waste. For more information on this site, please see [15, 16].

A lot of monitoring techniques have already been applied [17,18] on Olkiluoto, during the planning and construction phase as well as for implementing IAEA safeguards. Among the remote sensing techniques, SAR imagery show some advantages for monitoring nuclear sites in northern areas compared to optical data as they operate under almost all weather conditions and independently of the sunlight, i.e. time of the day.

QuickBird multispectral image, true colour, 07.09.2007 (Credit: DigitalGlobe)

TerraSAR-X, HH/VV, Stripmap mode, 21.10.2007 (Credit: Infoterra)

Figure 2: Olkiluoto Peninsula (Finland) in 2007, seen by QuickBird (left) and TerraSAR-X (right).

Based on a TerraSAR-X spotlight stereo pair, acquired in 2008 at November 31 with an incidence angle of 22.13° and a spatial resolution of 3.3 m, and at December 1 with 42.02° and 1.8 m resolution, a DSM was extracted through radargrammetry. Different approaches were tested: For speckle reduction, no, Median, Gamma and Lee filter were applied. For geometric correction, internal Ground Control Points (GCPs) from the input data were used as well as from an additional DEM derived from topographic maps [19]. The best results were achieved when no filtering but additional GCPs were employed (Figure 3). The values for the surface heights (i.e. including buildings, trees and other objects on the surface) amount around 20 m (green) and increase up to 40 m (yellow) and 60 m respectively for some outliers (red). For the accuracy assessment, some measured terrain heights were compared with the calculated surface heights. The height differences correspond approximately with the vegetation heights. The DSM, however, still shows some holes where due to insufficient coherence no heights could be calculated. The decorrelation from one day to the next is connected with the different roughness of the vegetation cover the two acquisition times.

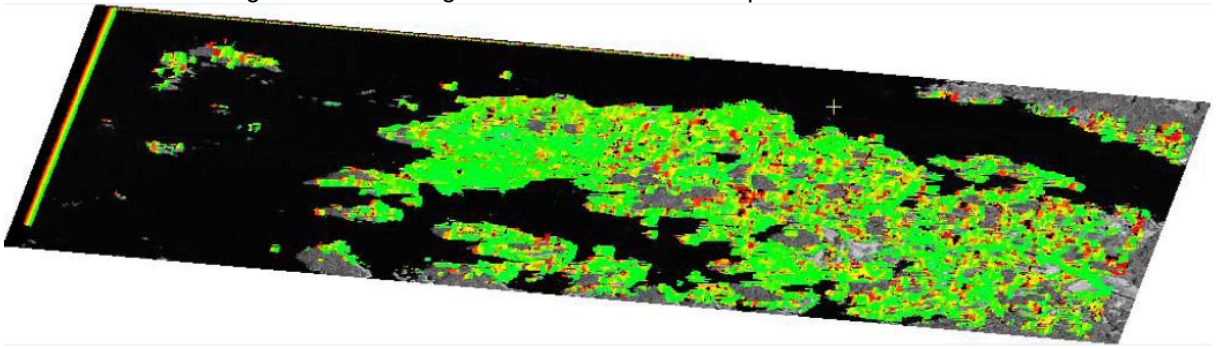


Figure 3: Digital surface model generated from a TerraSAR-X stereo pair (Credit: Infoterra).

4.2 Non-coherent change detection

For experiments on non-coherent change detection, two TerraSAR-X Spotlight scenes over the URENCO site in Gronau [20] were available(Figure 4, left), one acquired on November 4, 2007 with single polarization (HH) and on November 12, 2008 with dual polarization (HH/VV) (Figure 4, right)

QuickBird multispectral image, true colour, 11.09.2006
(Credit: DigitalGlobe)

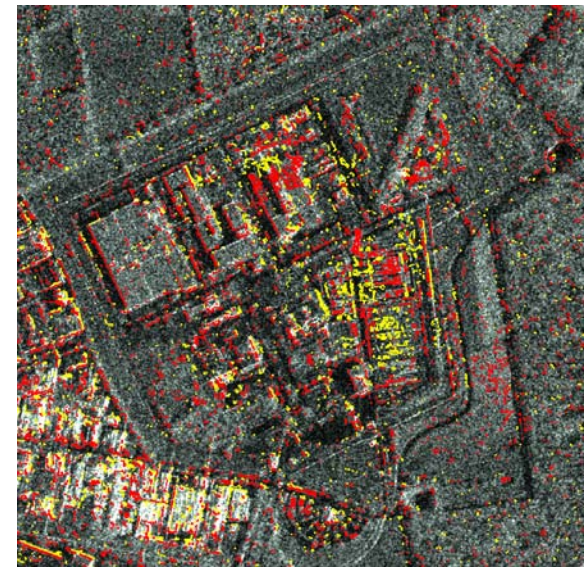
TerraSAR-X, HH/VV, Spotlight Mode, 12.11.2008
(Credit: Infoterra)

Figure 4: Urenco site in Gronau, seen by QuickBird (left) and TerraSAR-X (right).

Two different techniques were applied: A simple overlay of the three bands HH (2007), HH (2008) and VV (2008) in a Red-Green-Blue (RGB) colour composite (Figure 5, left) and the amplitude ratio of the bands HH (2007) and HH (2008) (Figure 5, right). In either case, a 5x5 Kuan filter was applied for speckle reduction.

The interpretation for the RGB overlay (Figure 5, left) is quite simple: All red- and cyan-coloured pixels indicate changes. Red pixels indicate a much higher backscatter intensity in 2007, the cyan ones in 2008. In the Gronau example, we have a lot of changes in the tails and feed storage area. Here, metallic UF6-cylinders were apparently moved or relocated within the storage area between 2007 and 2008. Other changes are related to the construction of the UTA-2 plant in this period of time.

The result based on the amplitude ratio (Figure 5, right) gives all positive (yellow) and negative changes (red) from +/- 5 dB. By this technique, also the changes connected to the storage area and the UTA-2 plant were detected but beyond that also a large number of false alarms. As they are presumably due to misregistration errors, an increase of the threshold would not better differentiate between the change and no-change pixels. Rather, more effort has to be spend on an accurate image-to-image registration and possibly also on the adaptive filtering.



Amplitude ratio of HH (2007) and HH (2008), negative changes < -5 dB (red), positive changes > 10 db (yellow); background: HH (2008)

Colour composite of HH (2007) in red, HH (2008) in green and VV (2008) in blue

Figure 5: Non-coherent change detection using a simple overlay (left) and the amplitude ratio (right).

5. Conclusions and future work

Non-coherent change detection provides a quick overview on the surface changes between two acquisition times in case the input scenes are registered properly. In order to derive detailed information, coherent change detection techniques are probably better suited.

The DSM extracted through radargrammetry does not yet provide satisfying results, but also here, there are some methodological options for improvements. In comparison to radargrammetry, InSAR is expected to provide DEMs with a much higher spatial resolution and precision. However, although less accurate than optical photogrammetry and SAR interferometry, radargrammetry can be applied in areas with extreme weather conditions where low cost and low resolution data are needed.

In areas with low coherence values due to vegetation cover and/or too large spatial or temporal baselines, some SAR analysis techniques for 3D information extraction and 2D/3D change detection

show only limited results or even fail. Thus, methodological improvements are needed with regard to the application of techniques using the coherence between SAR scenes acquired over an area at different acquisition times, such as CCD, (D)InSAR and radargrammetry.

But also technical innovations can enhance the chances for applying SAR imagery analysis in terms of safeguards purposes, for instance the availability of single-pass or dual-pass interferometry without temporal decorrelation. In this regard, the TanDEM-X (TerraSAR-X Add-oN for Digital Elevation Measurement) mission, scheduled to be launched in fall 2009, has promises. By this mission, a second, almost identical spacecraft will fly together with TerraSAR-X in a closely controlled formation with typical distances between 250 and 500 m. In this way, the temporal correlation between two SAR images over one area will be minimised.

As for vegetated surfaces, the application of L-band data being able to penetrate the aboveground biomass could be another solution. Today, only PALSAR onboard ALOS provides L-band data from space in 10m resolution. The European TerraSAR-L mission was intended to complement the X-Band TerraSAR-X satellite with a fully polarimetric L-band SAR with 5m spatial resolution. For the time being, the TerraSAR-L programme has been stopped, but hopefully the European Space Agency (ESA) will continue the mission one day.

Besides optical imagery, POL-SAR techniques can assist change detection studies and the identification of imagery indicators/signatures for the NFC processes, once full polarimetric SAR data will be available in the spotlight mode. Preliminary results based on dual polarimetric TerraSAR-X imagery already showed promising results

More advanced techniques like Polarimetric SAR Interferometry (PolInSAR) combining both POL-SAR and InSAR might also have potential for safeguards applications. Future case studies using polarimetric interferometric SAR data from Radarsat-2, ALOS PALSAR or TerraSAR-X will also focus on the PolInSAR advantages.

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LIMES: Satellite Imagery Analysis and Information Management for Treaty Monitoring

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Abstract:

LIMES (Land/Sea Integrated Monitoring for European Security) is a FP6-funded project which aims at developing satellite-based services for different security-related applications such as maritime, land & border surveillance and emergency response. LIMES started in December 2006 and will run until early 2010. The first half of the project has been concluded with a number of service demonstrations during the summer 2008.

LIMES contains a work package focused on Treaty Monitoring, which has the objective to provide an integrated platform supporting the image analyst in verifying treaty compliance. The main aspects addressed by the work package are:

- *increased automation of the image processing workflow, in particular in the areas of object-based change analysis, 3D information extraction and processing of radar imagery*
- *improved information management using a GIS-based platform capable of integrating data and documents from multiple sources and time-frames, including satellite imagery, site modeling, open source information, reports, etc*

A demonstration was carried out during the summer 2008. The scenario was to monitor and analyse the Finish nuclear site Olkiluoto since the beginning of the EPR construction in 2004 using satellite imagery and Open Source information. The demonstration and platform validation was performed at the European Satellite Centre (EUSC) and the results were presented to a number of potential users including IAEA and DG-TREN.

The paper presents the achievements of the Treaty Monitoring work package and in particular the results of the Olkiluoto demonstration and also discusses the planed activities for the remainder of the project.

Keywords: GMES; LIMES; treaty monitoring; earth observation; GIS; data processing and integration

1. Background

GMES (Global Monitoring for Environment and Security) is a European initiative for the implementation of geo-spatial information services dealing with environment and security [1]. It supports decision-making by both institutional and private actors concerning either new regulations to preserve our environment or urgent measures in case of emergencies and security threats. In order to take decisions, it is necessary to *anticipate*, *intervene* and *control*. GMES integrates these functions by assembling the information received from Earth Observation (EO) satellites and ground based information in a reliable, valid and compatible manner and will make them available for user friendly exploitation. The services will be used by environmental agencies, local, regional, national and international authorities, civil protection organisations, etc. GMES is in its implementation phase and the objective is to gradually develop and validate a number of pilot operational services, based on selected R&D projects extending and strengthening the current actions.

LIMES (Land/Sea Integrated Monitoring for European Security) is a FP6 Integrated Project funded by the EU [2]. It aims at the development of pre-operational GMES services to support security management at EU and global level. LIMES services are clustered in three groups:

- Maritime surveillance including open, coastal water and sensitive cargo surveillance.
- Humanitarian relief and reconstruction including services that cover the whole crisis cycle (disaster preparedness, operational support and support of reconstruction)
- Land and infrastructure surveillance including land border monitoring, critical infrastructure surveillance, support to event planning and Treaty Monitoring. The services are based on the capacity of Very High Resolution satellites, used in conjunction with medium to high resolution data and aerial imagery, to enable critical 4D spatial analysis of updated reference data.

LIMES started in December 2006 and has a duration of 42 months.

This paper describes the activities and objectives of the Treaty Monitoring work package. The work package is focused on the Non-Proliferation Treaty (NPT) and aims to provide an integrated framework and platform supporting Non-Proliferation image analyst in the verification of the NPT. It involves the following partners:

- European Commission Joint Research Centre (JRC), Italy
- Technische Universität Bergakademie Freiberg (TUBAF), Germany
- European Union Satellite Centre (EUSC), Spain

The Commissariat à L'Energie Atomique (CEA), France, contributes their expertise in SAR processing within the framework of a research collaboration.

The workpackage carried out a platform demonstration after the first 18 months. The test area was the Finish NPP Olkiluoto, where the first European Pressurized Reactor (EPR) is currently under construction.

2. Objectives

The developments under the Treaty Monitoring work package are targeted at the image analyst in the context of Nuclear Non-Proliferation who has the responsibility of collecting, managing and evaluating satellite imagery - often in conjunction with data from other sources - and extracting Non-Proliferation relevant information. The analyst generates a report on a country or location of interest and delivers it to the final user.

Following recent developments in Non-Proliferation the analyst is faced with new and increased challenges, as for example the detection of clandestine nuclear activities and the assessment of an increasing amount of multi-type information. Although satellite imagery is already an important tool in Nuclear Non-Proliferation, current applications rely heavily on visual interpretation with little use of automated processing. Furthermore, current analysis tools usually provide an isolated view on satellite imagery with poor integration of collateral data, such as Open Source information, GIS data, internal databases, reports, etc [3, 4].

In the near future, new satellite sensors (very high-resolution optical and radar imagery) will further increase the number of possible applications in Nuclear Non-Proliferation and, therefore, also the amount of data to be processed. Hence, the Treaty Monitoring Service aims at providing a framework supporting the image analyst in the forthcoming challenges.

The system includes the following features, thus contributing to the analyst's efficiency and effectiveness:

- Automatic change detection based on VHR optical data. An integrated object-based image classification helps differentiating various types of changes (e.g. vegetation changes or changes due to construction works). A user-friendly change viewer allows the analyst to easily assess the detected changes with respect to their relevance for Non-Proliferation.
- Support for the identification and localization of a site of interest through image classification techniques applied to medium resolution imagery. The classification highlights potential sites of interest for further investigation by the analyst.
- 3D site model generated semi-automatically from stereo satellite imagery to improve the analysis and visual interpretation of the site buildings and infrastructure. The 3D model can be used to detect structural changes by comparing it to a reference model obtained from an earlier stereo image or from in-situ measurements (i.e. ground-based laser scanning).
- The NPT Monitoring platform incorporates a tool that detects Non-Proliferation relevant anomalies in VHR SAR (Synthetic Aperture Radar) by analyzing series of interferrograms taken at different instants in time.
- Integrated, multi-source analysis of a nuclear site: the image analyst uses VHR satellite data and collateral information to identify the nuclear activities on the site or evaluate the status of a facility. The main entry point to the system is a geo-browser that provides accurate geo-spatial information. It allows augmenting the geographic features with additional data, such as non-spatial information which can be linked and viewed in HTML browser or large, high-resolution 3D models which can be opened and analysed in a dedicated 3D software. All information can be time-stamped allowing a 4D analysis of a site. The system is server-based, i.e. the information is shared between all authorized users while implementing security requirements through appropriate access control.

3. Platform Description

This section provides technical details of the main components of the Treaties Monitoring platform.

3.1. Change Detection and Analysis

A software tool has been developed to automatically detect and visualize changes in multi-spectral optical satellite imagery. The software (called ChangeView) is based on the Iteratively Re-weighted Multivariate Alteration Detection (IRMAD) method for statistical change detection, which is a proven and very effective change detection method for optical images [5,6,7,8]. It takes two co-registered multi-spectral images as input without the need for any further user-interaction or parameter tuning. The algorithm runs fast even on very large datasets and reliably produces the corresponding change map. The changes are colour coded according to their type, i.e. seasonal vegetation changes are coded differently from changes due to construction activities. The IRMAD method is also used for automatic radiometric normalization of images where the no-change pixels are first identified and used to normalize the images [9].

Some of the changes (e.g. seasonal vegetation changes) might, however, not be relevant to Non-Proliferation. Therefore, a visualization tool supports the analyst in assessing the resulting change map: it provides an overview of the colour-coded change map thus highlighting potential areas of interest. When the user clicks on a particular point in the change map, the tool displays the original imagery of the corresponding area (at both instants of time) and the resulting change map in full resolution. Thus the analyst can quickly identify all changes which are of Non-Proliferation relevance (see Figure 6).

The platform also incorporates change analysis using object-based image classification to further analyse the change map and automatically identify Non-Proliferation relevant changes (see Figure 7). The classification is based on a rule set, which contains both geometric and radiometric rules for automatically identifying objects in the image [10,11,12]. The rule set is formulated by an experienced user; it is generally transferable from one image of a site to another provided that the images are radiometrically normalized. However, it is usually not transferable from one site to another. The speed for creating the rule set depends on the quality of the data and the experience of the user.

The change analysis is most useful for sites which are regularly monitored, so that the time invested in the creation of the rule set is compensated by the time gained during analysis. Additional work for enhancing the workflow and user-friendliness is currently on-going.

3.2. 3D Information Extraction

Digital Surface Models (DSMs) are extracted from stereo satellite imagery for the purpose of improving the ortho-rectification of the VHR satellite imagery. The Treaty Monitoring workpackage made use of the RSG software package provided by Joanneum Research [13], who is also a partner of the LIMES project. The software implements the following stereometric processing workflow (Figure 1):

- **Sensor modelling**: The mathematical model of the sensor allows mapping a point from ground into the image and vice versa. Based on ground control measurements and/or tie-point measurements, sensor parameters can be optimized in a least squares manner.
- **Image matching:** The core procedure of the stereometric processing is the measurement of corresponding points in the stereo image pair, which is done by automated image correlation. The basic output is the geometric differences between the corresponding points which are known as *parallaxes* or also *disparity vectors*.
- **Point intersection**: the point intersection converts the 2D coordinates of the matching pixels into the 3D Cartesian coordinates of the corresponding ground point.
- **Interpolation**: The input to the final DEM/DSM generation procedure is given in ground coordinates being stored in raster file format (output of previous intersection). These ground coordinates represent irregularly distributed locations on the ground, for which the terrain height is known. Respective interpolation techniques are either a linear interpolation based on nearest neighbour re-sampling or an areal interpolation approach.

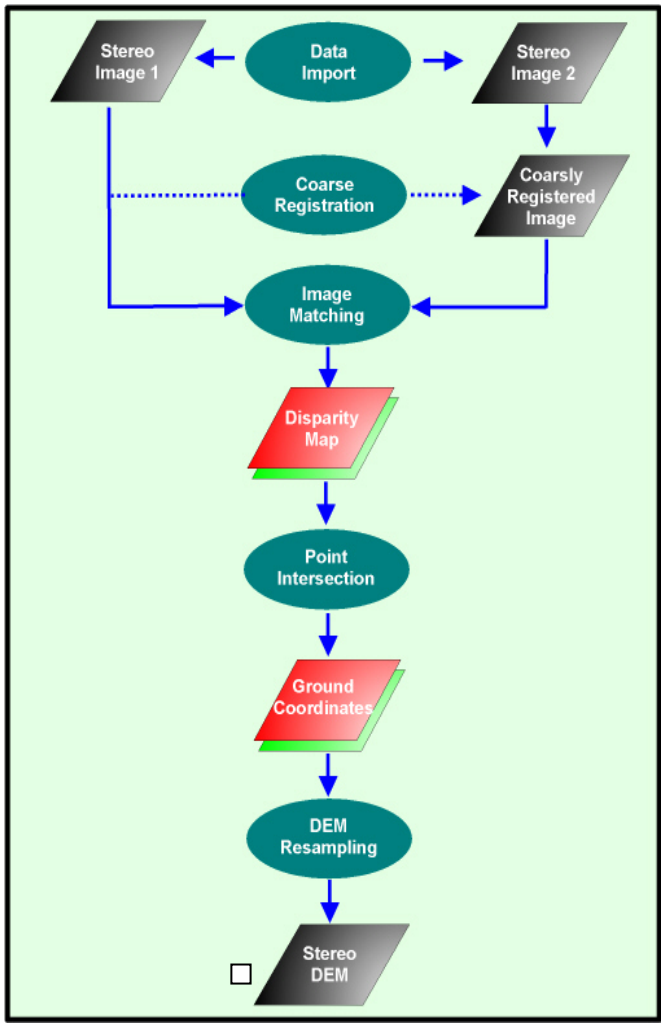

Figure 1: Stereometric processing workflow implemented by the RSG software package

Standard techniques, which are normally used for the creation of Digital Terrain Models from lower resolution imagery, often produce unsatisfactory results for the Non-Proliferation application [14]. For example, man-made structures with sharp contours (such as nuclear facilities) are blurred in the resulting DSM. Also, depth continuities and occlusions as they appear in VHR imagery generate mis-matches thus producing erroneous results. The standard methods have been improved within the LIMES project in order to cope with these challenges.

An additional usage of DSMs is 3D change detection, which is based on a simple comparison of two DSMs extracted at different instants of time [15]. A successful detection requires two DSMs of sufficient resolution and accuracy, which in turn can only be obtained from few stereo sensors, for example IKONOS or Worldview. Also, the processing workflow of 3D change detection is more complex and time-consuming as compared to 2D optical change detection. However, 3D change detection has the advantage that it directly detects changes relevant for the Non-Proliferation applications, i.e. changes in the geometry of the site which might be induced through construction activities. Irrelevant changes, e.g. due to seasonal changes are not detected. As both, input data and processing workflow, will improve in the near future, 3D change detection can become an important tool for the image analyst.

Last but not least, an important application of the DSM is for creating 3D models, which can be used for presentations, visual interpretation and generating fly-throughs.

3.3. SAR Processing

SAR (Synthetic Aperture Radar) interferometry exploits the phase difference between two SAR images taken from different viewpoints and allows the generation of digital elevation models, displacement maps and coherence maps. SAR provides high quality images in all weather and time conditions. Due to their complex nature, SAR images are not well suited for visual interpretation by the human eye. Consequently, automated processes for detecting anomalies are required.

The Treaty Monitoring platform uses the CIAO software package, developed by CEA, for detecting anomalies by the means of analysing series of interferograms taken at different instants in time [16]. The analyst first collects the radar images for a given area (different radar satellites eventually) and then uses the software to identify possible coherence images. Dark areas in the coherence image indicate changes between two images and the bright areas indicate possible infrastructure which did not change between the two image acquisitions. With this anomaly detection completed, the analyst can investigate suspicious areas further using another coherence image or optical images that provide greater resolution. Figure 2 shows a schematic view of the anomaly detection and Figure 3 gives an example of the process applied to an industrial port area.

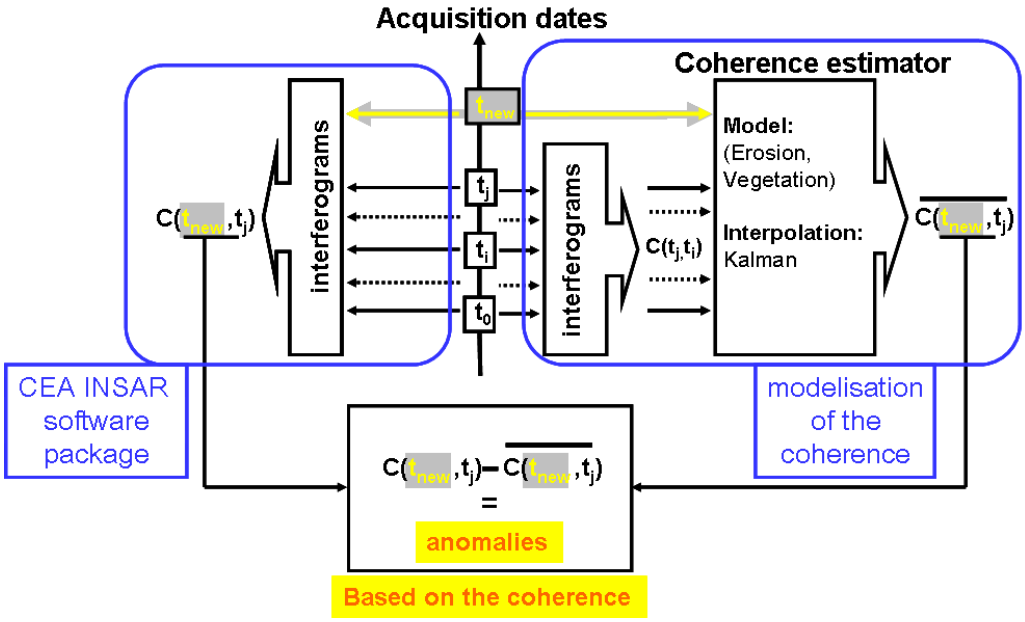

Figure 2: Principle of anomaly detection from coherence image

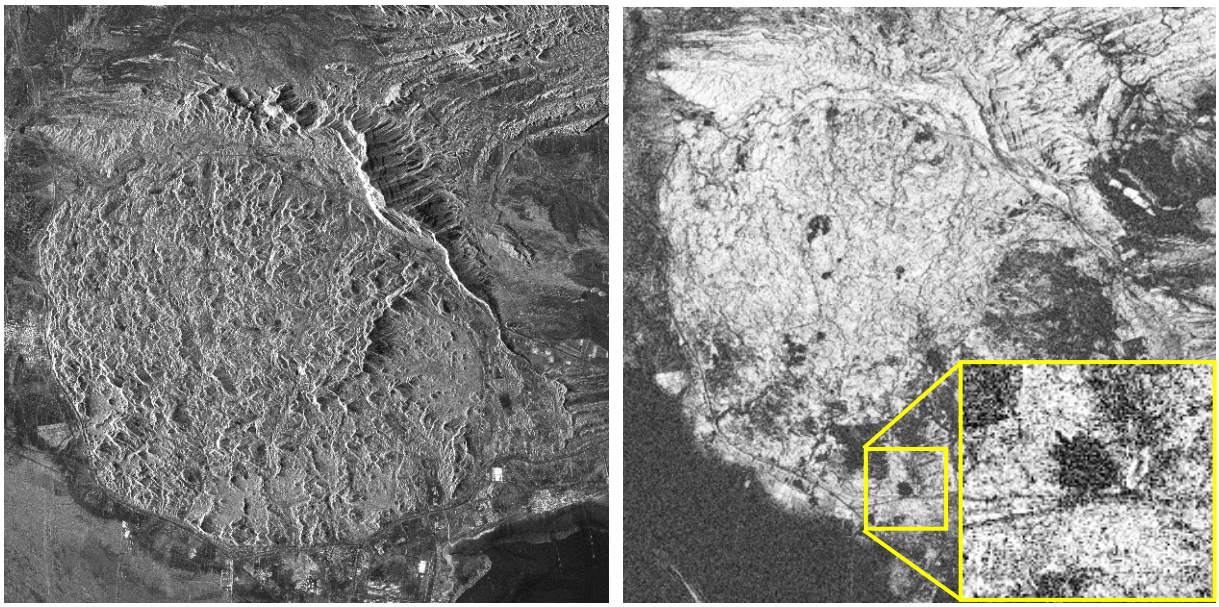

Figure 3: The left image shows a SAR image of an industrial port area. The right image shows the coherence map of the same area obtained from two SAR images. The yellow square highlights a detected anomaly.

3.4. Information Management and Integration

A core objective of the Treaty Monitoring workpackage is to provide an integrated platfrom to the Non-Proliferation image analyst. In practice this means, that the analyst should have a central point of access, which allows to

- Retrieve, view and analyse all available (spatial and non-spatial) information for a given site, including satellite imagery, GIS information, external databases and collateral information.
- Access dedicated analysis software performing specialist tasks, such as the change analysis, SAR and 3D processing tools described above. Any results obtained from the tools (e.g. a resulting change maps) should be stored for later retrieval in the integrated platform.

The integration platform is based on a standard three-tier architecture (database, application server and web client) as illustrated in Figure 4. It is implemented using common industry standards, therefore, it can easily be integrated in existing IT infrastructures, in particular with respect to existing security structures. All information is stored in a central database, hence it can be shared within the organisation on a need-to-know basis.

In order to support information of different types (both spatial and non-spatial) the platform incorporates three independent pillars each serving a particular purpose: a *geographic information system*, a *wiki system* and a *document repository*:

- The Geographic Information System (GIS) provides an intuitive map-based interface the user. It allows to store, retrieve and visualise all spatial information including multi-temporal satellite imagery, vector information produced through manual or automatic analysis and other GIS information. The focus is on user-friendliness and scalability, e.g. large (gigabyte) images are served as image pyramids for easy navigation. Each feature in the geo-database is context-sensitive, i.e. it can be selected from the user interface and cross-linked with information in the other pillars.
- The objective of the Wiki system is to capture unstructured, tacit information available in the organisation, which is needed to support the Non-Proliferation analysis. Each feature in the geodatabase (e.g. a particular facility) can have a corresponding wiki page containing relevant information or previous analysis. However, the Wiki goes further and might contain supporting information, for example pages regarding relevant technologies, organisations, treaties, regulations, etc. Wikis are most known as Internet applications (e.g. Wikipedia), where the huge number of contributing users ensure reliable and exhaustive content. However, the

potential of Wikis is also increasingly recognised in Intranet environments with a smaller number of users, e.g. in corporate intranets or for sharing information within intelligence communities, as they provide a low level-of-entry for contributing and sharing information [17,18]. A prominent example is Intellipedia, which is an online system for collaborative data sharing used by the United States intelligence community [19, 20]

- The document repository is a central archive for all relevant background documents, in particular for Open Source documentation collected from the Internet. Open Source information is becoming increasingly important to trigger, guide and support imagery-based analysis. The document repository provides a structured archival and easy retrieval of the documents.

The three pillars are integrated on the data level through explicit cross-links and a common data dictionary and on the interface level by providing hyperlinks to relevant information, e.g. the user can directly navigate from a spatial feature in the geo-browser to the corresponding wiki page and from there to all relevant source documents.

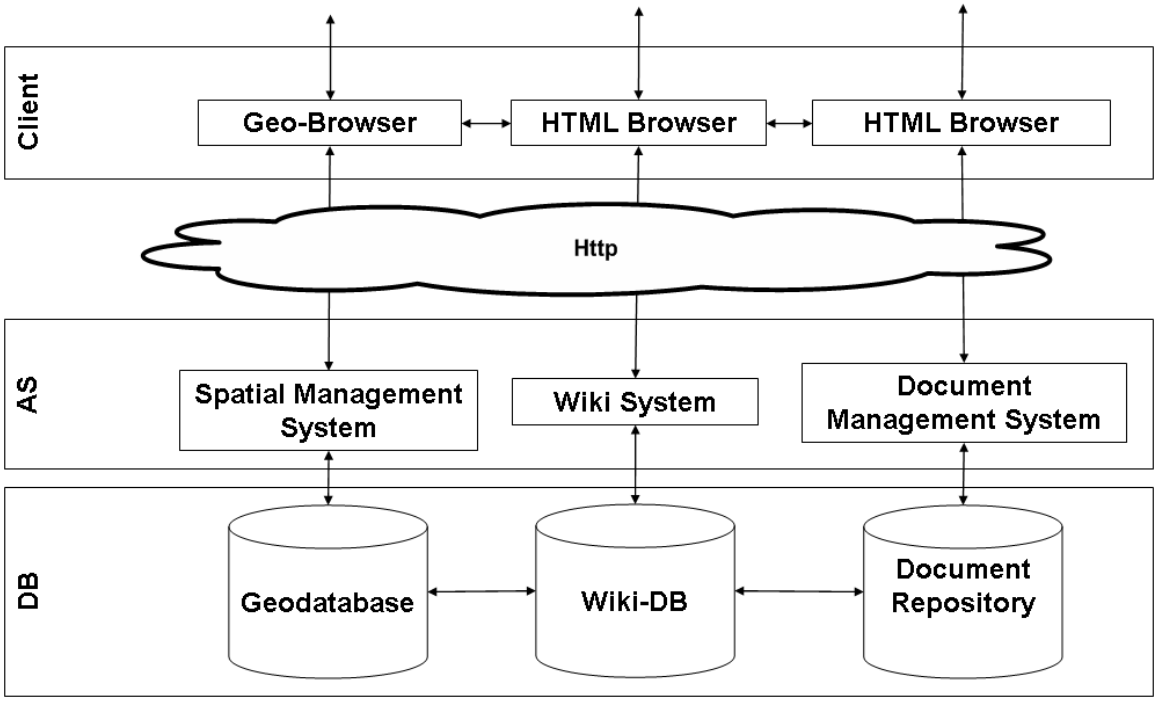

Figure 4: High-level architecture of the Treaty Monitoring integration platform. The system integrates a Geographic Information System (left), a Wiki (centre) and a document repository in order to support spatial and non-spatial information.

4. Platform Demonstration

The Treaty Monitoring platform was demonstrated to potential users in July 2008. The objective of the demonstration was to present the developments to interested stakeholders and to obtain their feedback for guiding further developments during the second phase of the project. For the purpose of the demonstration, a scenario was defined incorporating three common tasks faced by the Non-Proliferation analyst:

- **Site Localisation**: The objective is to support the image analyst in localizing a nuclear site within an area of interest, i.e. given only rough coordinates.
- **Site Analysis**: The Site Analysis covers the initial baseline analysis of a nuclear site.
- **Site Monitoring**: The objective is to monitor a nuclear site over a longer period of time (e.g. several years) in order to detect relevant changes, e.g. construction works or other activities of interest.

The Finish NPP at Olkiluoto was selected as test site for the service demonstration. The motivation for the choice was that i) it hosts two existing nuclear reactors and a third reactor is currently under construction, ii) it was possible to obtain ground-truth information and validate the outputs of the analysis and iii) the site authorities (STUK) and operators (TVO) were very collaborative in supporting LIMES activities. Archived satellite imagery reaching back to 2002 as well as new imagery from autumn 2007 and spring 2008 was acquired, thus simulating a continuous monitoring of the site over six years. The data used for the demonstration include

- VHR optical satellite imagery (Quickbird: 2002, 2005, 2006 and 2007)
- VHR optical stereo imagery (EROS : 2007, Ikonos : 2008)
- Landsat imagery
- SAR imagery
- Open Source information including documents, maps, images and videos.
- Ground information: GPS

Figure 5 shows snapshots of some of the multi-spectral VHR imagery acquired over Olkiluto. Figure 6 to Figure 8 provide some examples of the results obtained from the automated image analysis tools. Figure 9 shows how the different types of information are brought together in a single integrated environment.

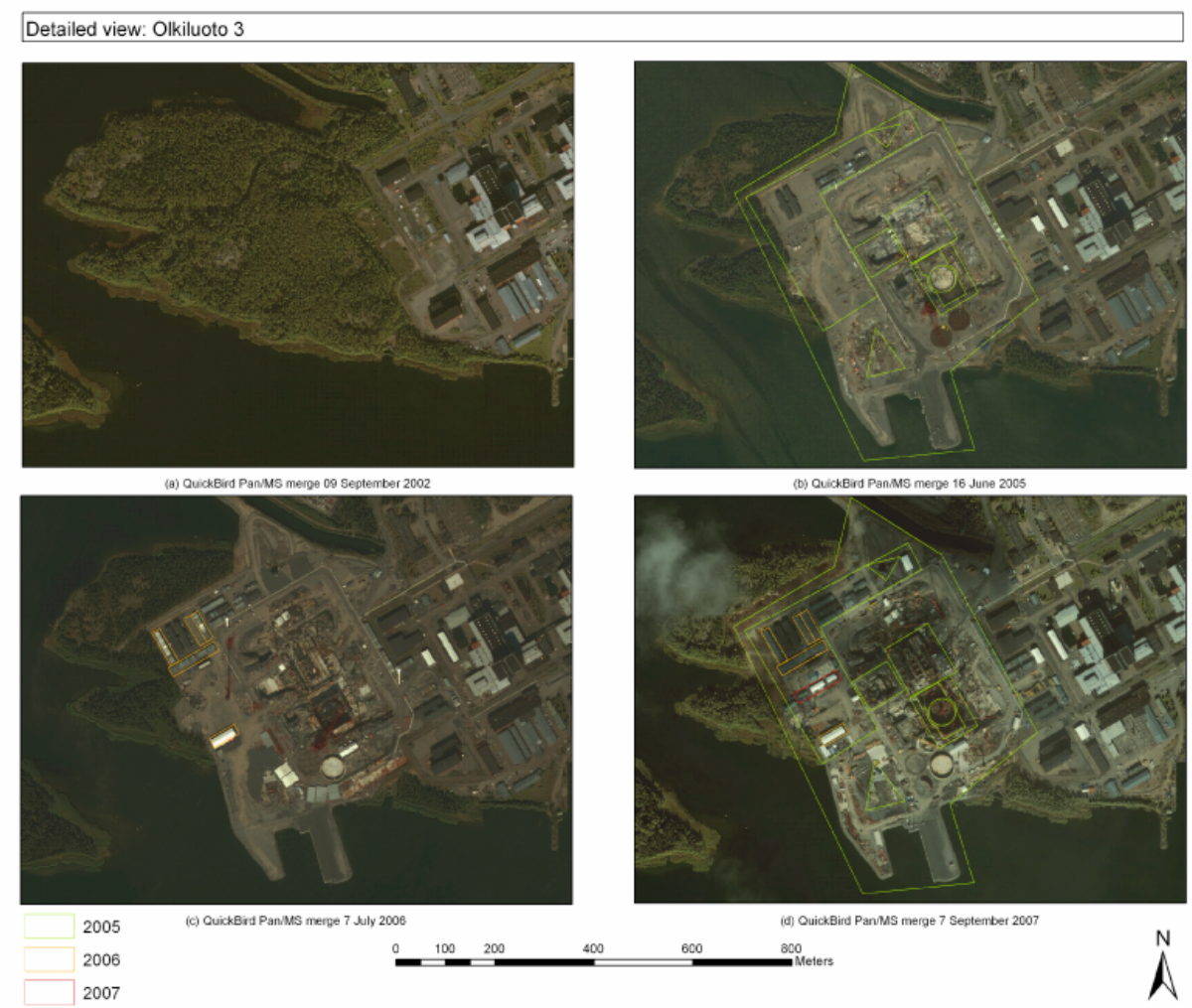

Figure 5: The four images show the site of the EPR reactor before the construction (2002, top-left) and at different points of the construction (2005, 2006 and 2006). The images include the annotations resulting from the standard, visual interpretation carried out by an image analyst.

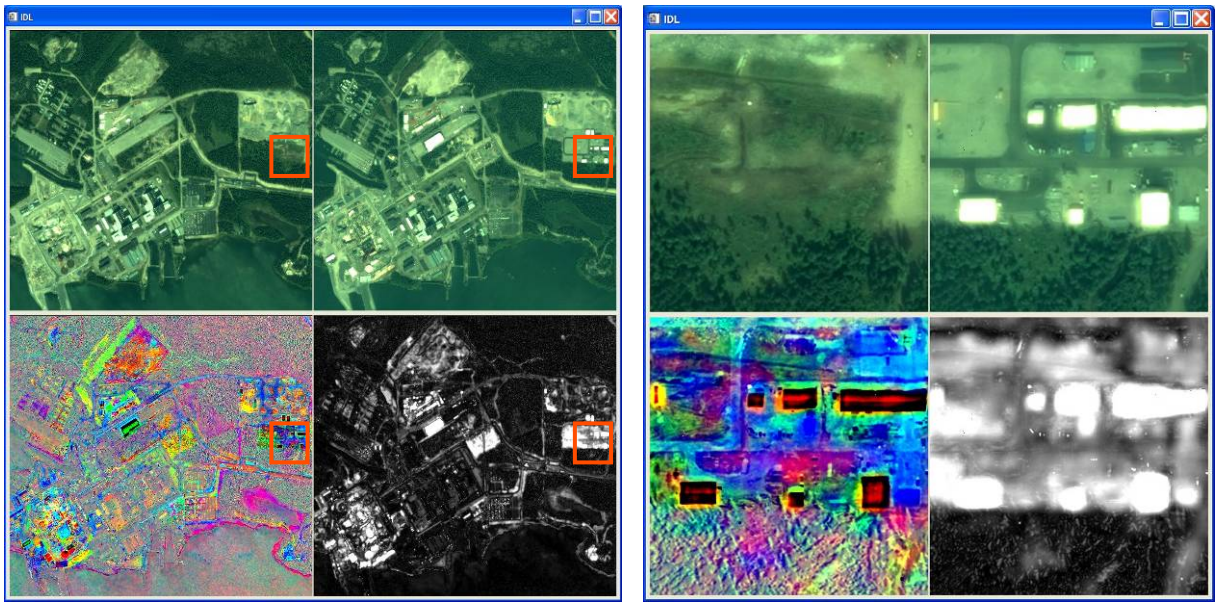

Figure 6: Snapshot of the TUBAF change visualisation tool: The left window displays an overview of the original Olkiluoto images taken in 2005 (upper left) and 2006 (upper right) as well as the resulting change map as colour coded MAD components (lower left) and absolut change intensity (lower right). The right window displays the same information in full resolution for a given area of interest, which can easily shifted by moving the red square in the overview window.

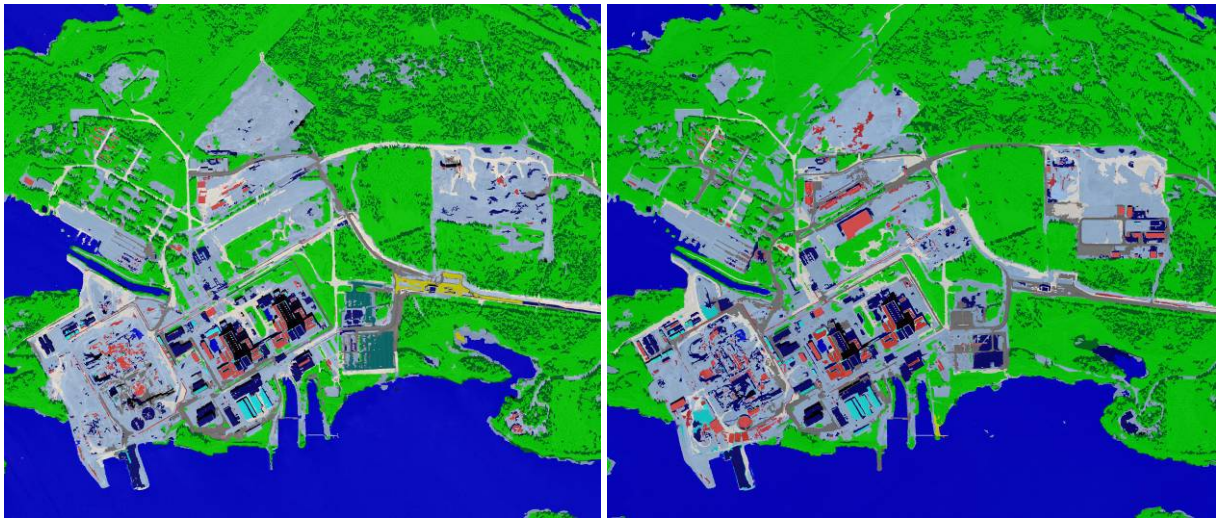

Figure 7: Result of applying the object-based image classification to the 2005 (left) and 2006 (right) Olkiluoto images. A semi-automated change analysis can be achieved by fusing the change detection and image classification components.

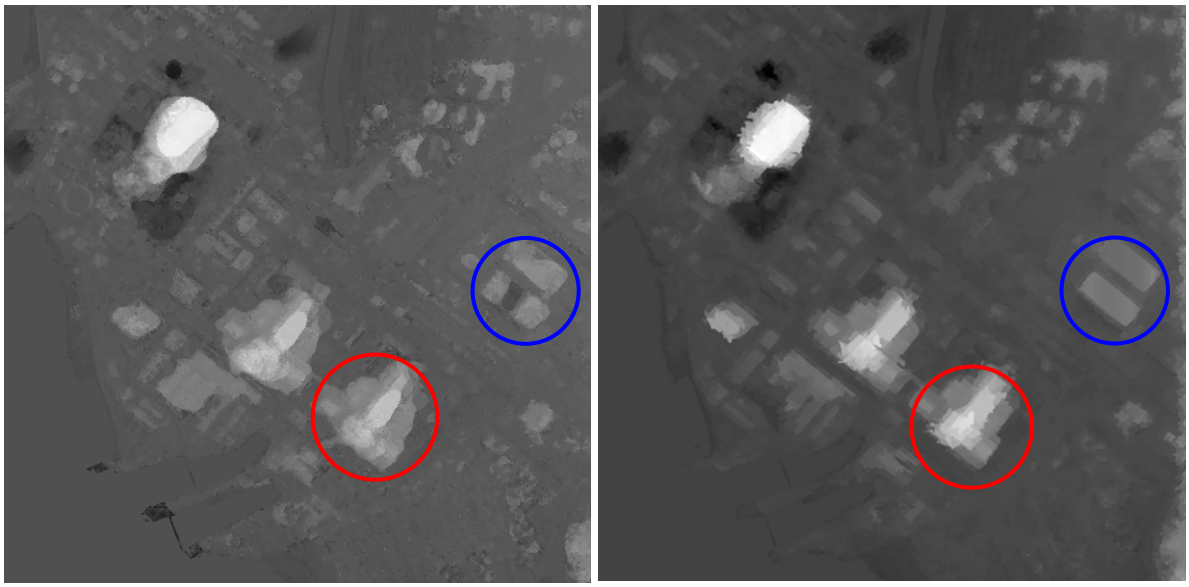

Figure 8: Digital Surface Maps (DSM) of the Olkiluoto site generated with the Joanneum RSG software from an Ikonos stereo pair (2008). The left DSM results from applying a standard method; the right DSM is generated using an advanced algorithm. The red circles highlight the improved building contours: in the left image the building (one of the existing power reactors) appears blurred; in the right image the contours are more accurate. The blue circles highlight an example of improved image matching: the left image contains errors induced by the depth discontinuities whereas the right image shows the correct results.

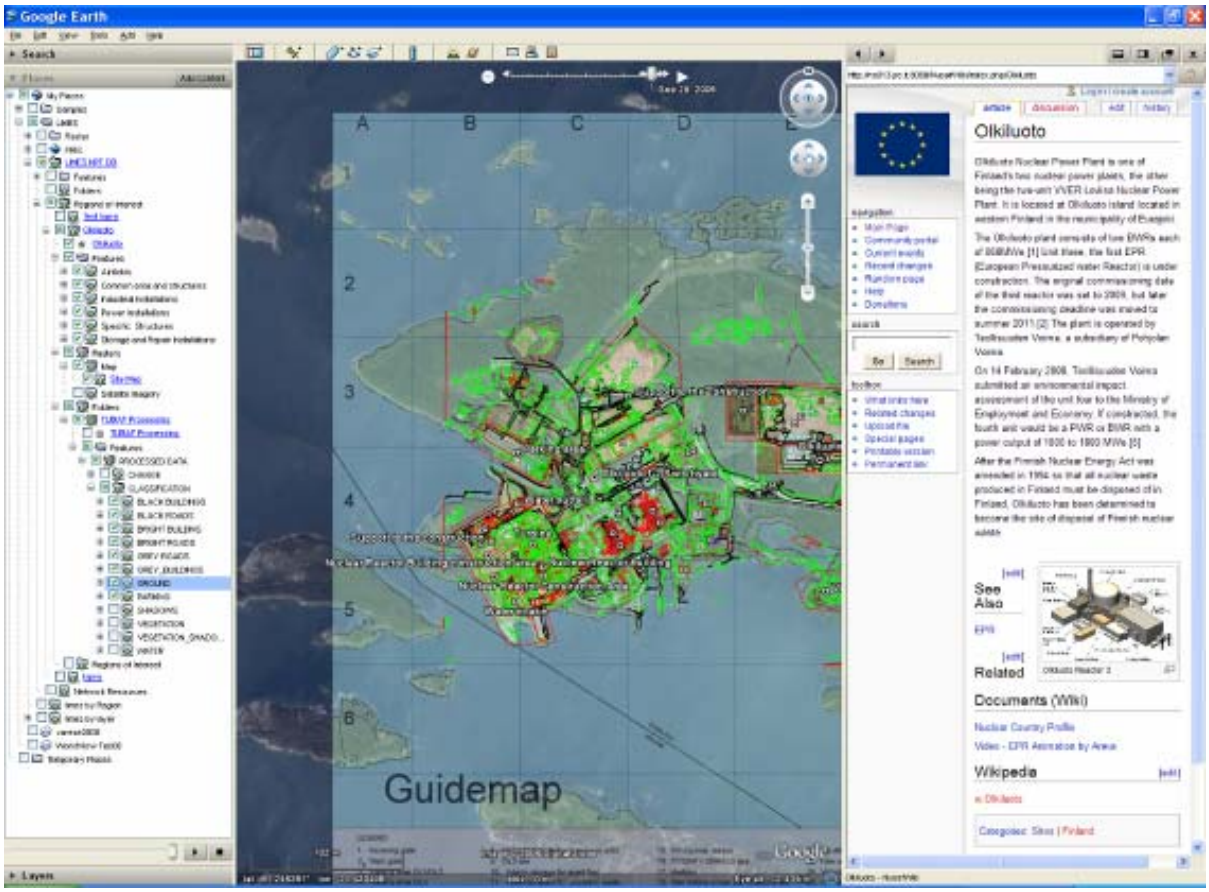

Figure 9: Snapshot of a geo-browser (Google Earth in this case), which is visualising the spatial and non-spatial infromation served by the integration platform. The spatial information (multi-temporal vector or raster information) is selected from the hierachical tree on the left. For each feature, related non-spatial information can be loaded into html browser on the right.

For the service demonstration, the platform was deployed at and tested by the European Satellite Centre based on the three scenarios described above. The results of the testing were presented to and discussed with other stakeholders in the nuclear non-proliferation monitoring community including IAEA and DG-TREN during a demonstration workshop.

The main feedback received from the testing at the EUSC and from the participants of the user workshop is summarized below:

- The main focus of the Treaties Monitoring workpackage (i.e. increased automation for image processing and improved information integration) are some of the most relevant issues for the Non-Proliferation analyst.
- The components either provide already added value to the Non-Proliferation analyst or will do so once they achieve the performance level that is foreseen by the project partners.
- The implementation (user-friendliness, functionality, reliability, etc) was judged positively considering the pre-operational status of the platform,
- The collaborative nature of the integration platform was subject to intense discussion. On one hand, collaboration and information sharing between analysts is considered important. On the other hand, it was noted that for some organizations, there may be difficulties to put all these features into practice. Influencing factors include: organization culture, its hierarchical structure and internal security constraints.
- Security considerations are of paramount importance in the area non-proliferation monitoring and hence should be reflected in all aspects of the Treaty Monitoring platform.

5. Summary and Outlook

The paper presented the activities and results achieved by the Treaty Monitoring work package during the first half of the LIMES project. The workpackage develops an integrated platform which aims to support the Non-Proliferation image analyst in the verification of Non-Proliferation Treaty compliance.

The work package partners bring together a number of semi-automated image analysis tools addressing some of the most critical issues for the Non-Proliferation image analyst, namely i) object-based change analysis using optical imagery, ii) automated analysis of complex SAR imagery and iii) stereometric 3D processing.

The tools support the analyst by drawing the attention to a particular anomaly which then needs to be further investigated by validating it with other information. Each tool provides pieces of information adding to the larger picture. Hence, there is the need for a single environment, which brings together all pieces and allows the analyst to have a view of the global picture. Therefore, the workpackage implemented an integrated platform allowing the analyst to access all spatial and non-spatial information required for a given task including the original raster and vector data, the results obtained from the image analysis, previous reports and other available collateral information, e.g. documents collected from Open Sources. The integration platform is conceived as collaborative environment, which means that it not only integrates information of different types and sources, but also allows sharing information between different analysts and groups within the organization.

The platform was demonstrated during a user workshop to a number of interested stakeholders (including DG-TREN and IAEA) by applying a typical Treaty Monitoring scenario to the Olkiluoto NPP in Finland. The feedback received from the demonstration guides the developments for the second project phase, which include further work on the 3D change analysis, SAR processing and information management. A final project demonstration is planed for October 2009.

Acknowledgements

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Detectability of clandestine nuclear reprocessing analysed by atmospheric krypton-85 modelling

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Abstract:

As project within the German support program to IAEA Safeguards, the detectability of additional ^{85}Kr sources using atmospheric transport modelling was investigated. ^{85}Kr is released into the air during reprocessing of spent nuclear fuel rods. Therefore it can possibly be used as indicator for the detection of undeclared plutonium separation. First, the global ^{85}Kr background produced by known reprocessing facilities from 1971 until 2006 was simulated with the atmospheric general circulation model ECHAM5 using annual emission data. The model results were evaluated by extensive comparison with measurements performed by the German Federal Office for Radiation Protection. Of particular interest for an assessment of the detectability of unknown sources is the background variability. The variability of concentrations is very high over central Europe, where the large reprocessing plants La Hague and Sellafield are located, and it is very low on the Southern Hemisphere, where no nuclear reprocessing takes place. The analysis of concentration time series on various time scales allows partly a distinction between fluctuations caused by the variability of thesources from variations due to atmospheric dynamics. Furthermore the detection sensitivity to a set of arbitrarily specified source locations is analysed with a lagrangian particle dispersion model. This, in combination with the location specific background variability, is giving first benchmarks on the capability of using ^{85}Kr for IAEA Safeguards on the NPT based on the Additional Protocol and its possible contribution for the verification of a future Fissile Material Cut-off Treaty. For that, sampling procedures impose high requirements on measurement technology, especially in terms of sample size in the field and cost effectiveness. Therefore the Centre for Science and Peace Research develops in another project a new spectroscopic ultra-trace measurement technique for ^{85}Kr in a magneto-optical trap, the so called Atomic Trap Trace Analysis.

Keywords: Environmental sampling; atmospheric transport modelling; krypton-85, reprocessing;

1. Introduction

In spite of the broad range of methods available to IAEA for safeguarding nuclear facilities and fissile materials, there are deficits in detecting undeclared activities. Therefore the Additional Protocol on the NPT was developed with more comprehensive and technically advanced methods against non-compliance. There are inspections on short notice (24 hours) possible and rigorous declaration obligations. Furthermore there is the possibility of environmental sampling given. Location specific samples are taken at the facilities directly, for example so called swipe samples collecting dust from the surfaces for spectrometric analysis in the laboratory. The possibility of wide area environmental sampling - i.e. sampling not in the direct vicinity of known facilities – is principally foreseen, but its application is not yet decided by the IAEA-Board of Governors. The project in the IAEA support program of the German Federal Ministry for economy and technology shall serve as feasibility assessment to prepare further IAEA decisions on using ^{85}Kr for the detection of reprocessing activities. The project follows a multiphase approach – phase I lasts two years. The subject of the first year described in this article was the analysis of the global ^{85}Kr background and investigation of the potential detectability of additional sources with atmospheric models.

2. Excursus: Atmospheric transport modelling

Due to the strong improvement of computing performance and the risen interest in climate research, atmospheric models became more and more sophisticated and allow for advanced tracer transport modelling in high resolution. For calculating atmospheric transport the knowledge of the wind fields and their development in time is necessary. There are to ways to acquire the meteorological data: The model can calculate the meteorology itself by solving prognostic equations for the variables describing the state of the atmosphere (also called online calculation) or it receives externally generated meteorological variables, e. g. reanalysis or prognostic data (online calculation). For the tracer transport calculations itself two approaches exist: The particle following lagrangian mode and the fixed grid box based Eulerian method. A further development of the lagrangian trajectories are plume dispersion models, which consider the turbulent diffusion of the plume. Such Lagrangian Particle Dispersion models are already applied in the Provisional Technical Secretariat of the Preparatory Commission for the Comprehensive Test-Ban Treaty Organization in Vienna. Besides the waveform based technologies for measuring time, location and strength of explosion events, measurements of radionuclides are necessary to detect radioactive debris from nuclear explosions. In the International Monitoring System (IMS) 80 radionuclide stations are planned (more than two third are finished). To get the meteorological relationship between possible source regions of the emission and the monitoring station(s) Atmospheric Transport Modelling is applied [2]. For the detection of nuclear tests radioactive xenon isotopes are very important, as they are likely to be released also from underground explosions. Over the combination of specific isotopic ratios it is possible to distinguish reactor and explosion sources from each other [3]. As the radioactive half-lives of the xenon isotopes of interest range from 9 hours to 12 days, the background from civilian isotope production facilities and nuclear reactors is soon decaying and the signal of a nuclear explosion is much stronger than the regular civilian emissions (But then also rapidly vanishing). The situation for using krypton-85 to detect reprocessing activities is even more challenging: As the radioactive half-live of ^{85}Kr is 10.8 years, it remains in the atmosphere - so there is a high background, and the signals of interest are small compared to industrial emissions.

3. Using ^{85}Kr as indicator for plutonium separation

The radioactive noble gas isotope ^{85}Kr is produced along with the plutonium in nuclear reactors. It remains bound in the nuclear fuel rods until reprocessing. When the fuel rods become dissolved chemically, the ^{85}Kr gets released into the air. Its radioactive half-life is 10.8 years. There are no other relevant sources of ^{85}Kr existing. Natural generation by cosmic rays is by more than six orders of magnitude weaker than the anthropogenic reprocessing sources. The operational releases of nuclear power plants and isotope production are small compared to the reprocessing emissions. As noble gas it is chemical inert and has low solubility in water. Therefore, its behavior in the atmosphere is easy to describe as there is no deposition to be considered. Over the decades of the weapons material production and civilian reprocessing the global content of ^{85}Kr continuously increased. Figure 1 shows the activity amount of ^{85}Kr in the atmosphere over the years since 1945. In former times, the US-military were supposed to use ^{85}Kr analysis for an estimation of the Soviet plutonium stockpiles and Frank v. Hippel did so for the public [4]. There was a case study in the 1980s investigating weekly ^{85}Kr measurements at various distances from the German reprocessing facility Karlsruhe leading to considerable detection probabilities without applying atmospheric models [5].

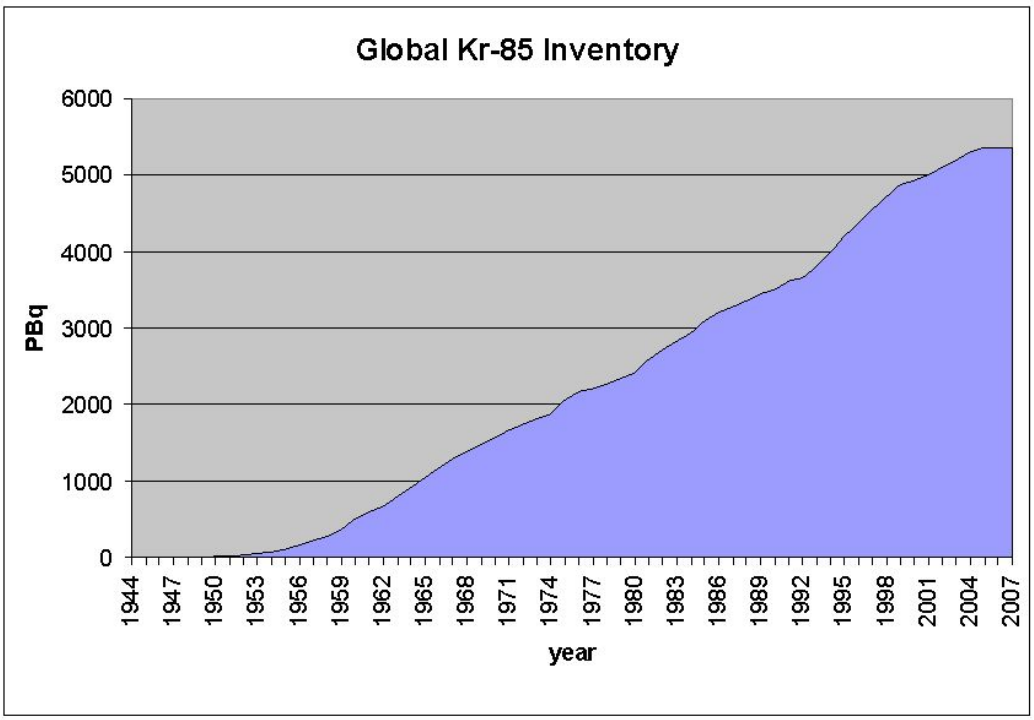

Figure 1 Global atmospheric content of ^{85}Kr from reprocessing as activity in PBq, (decay included)

4. Simulation of global ^{85}Kr background with ECHAM5

ECHAM5 is based on the global weather forecast model of the ECMWF and was partly developed at the Max Planck Institute for Meteorology in HAMburg. Thus, ECHAM5 is the fifth generation of the European Centre-Hamburg general circulation model. The prognostic equations of the dynamical core are calculated in a spectral representation. A detailed model description is given by Roeckner et al. [6]. ECHAM5 contributed to the scenario experiments used in the fourth assessment report of the intergovernmental panel on climate change. The tracer transport of the model consists of large scale advective transport, vertical turbulent diffusion, and cumulus convection. A new submodel for emissions, transport, and radioactive decay of ^{85}Kr was implemented into ECHAM5. After first tests the model was modified in a way that basic requirements for tracer transport as mass conservation and non-negative concentrations were fulfilled.

4.1. Emission inventory and simulation setup

The emissions from reprocessing plants were introduced with yearly constant data from a updated and extended version of the emission inventory created by Winger [7]. The used emission data are published on the world wide web sites of the Independent Group of Scientific Experts for the detection of nuclear weapons usable material production (www.igse.net) and are available as occasional paper of the Centre for Science and Peace Research [8]. The simulated time period was from 1971 to 2006. The spectral resolution was chosen to T63 (approximately 200 km) with 31 vertical levels - the time step length was 12 minutes. An initial tracer concentration field for Dec 31st, 1970 was derived from a former model run done by Katja Winger with ECHAM4 [7]. As in this former study a net loss of ^{85}Kr occurred, the initial tracer field was globally scaled up according to the expected global ^{85}Kr amount from the emission inventory. Temperature, surface pressure, divergence, vorticity, and sea surface temperature were constrained to ERA-40 reanalysis data with a relaxation method (so called nudging) on all levels. The model ran on a NEC-SX 6 high performance computer (vector machine).

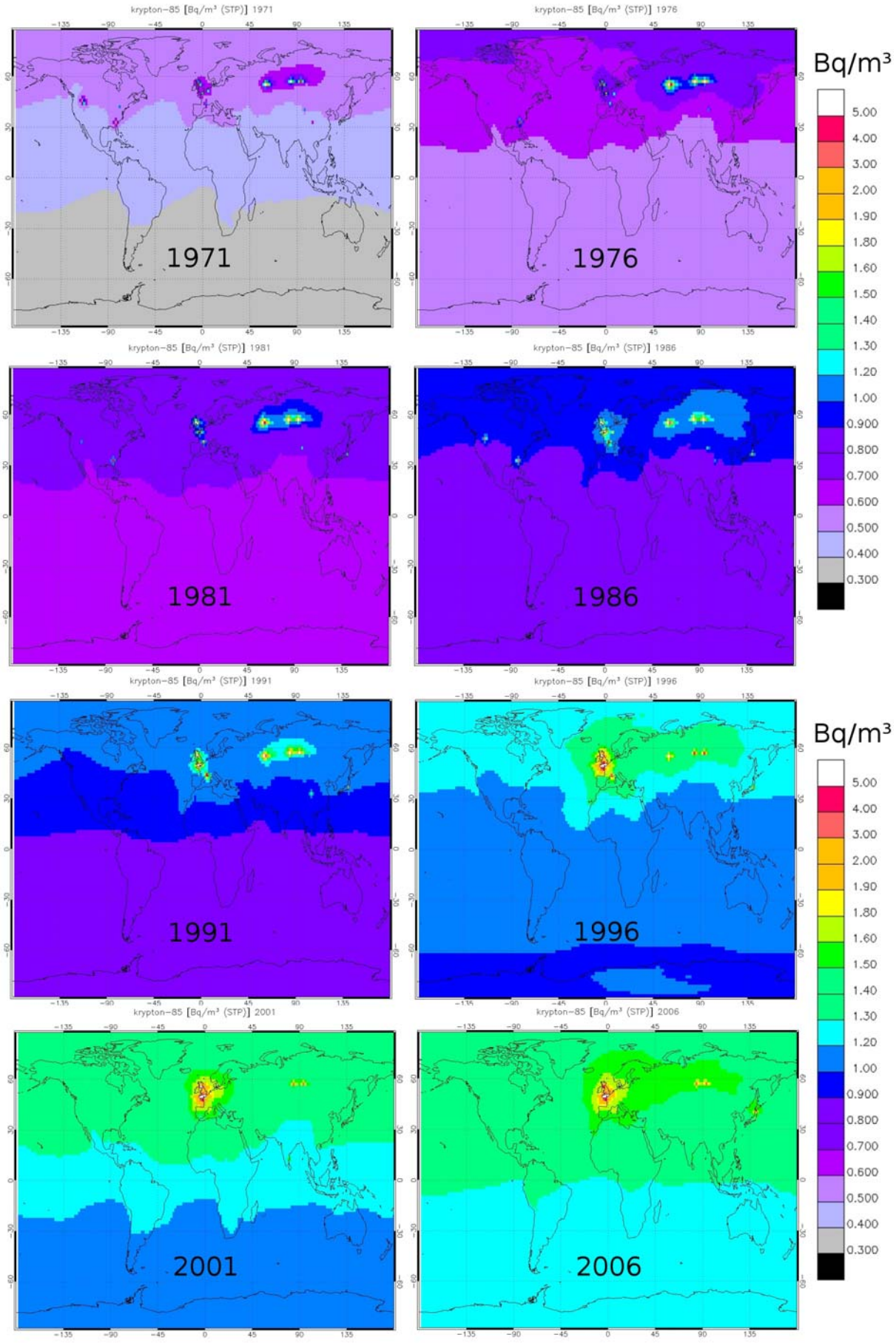

Figure 2 Distribution of modeled annual mean ^{85}Kr-concentration at surface 1971-2006 (every 5 y)

4.2. Results and comparison with measurements

Figure 2 shows the mean surface concentration distribution for exemplary years. The USA stopped reprocessing and the Russian sources weakened in the 1990s, but the European plant at La Hague doubled its emissions. In the figures the overall increase of concentrations over the years is clear - with a specific delay on the Southern hemisphere. ^{85}Kr is suitable to study the atmospheric interhemispheric transport. While the zonal air mass exchange takes place in orders of weeks, air needs about one year to cross the equator from mid latitudes. As there are no active reprocessing plants on the Southern hemisphere, the variability of the background is very low there. The highest variability can be found in the vicinity of strong sources. The emissions of La Hague and Sellafield dominate whole central Europe. When the new Japanese reprocessing plant Rokkasho becomes operational at full scale, it will be a source with comparable strength. For evaluation, the model results were compared with measurements provided by the German Federal Radiation Protection Office. The results showed a very good agreement between simulations and measurements. The large scale dynamics of the global distribution was excellently modelled. The measurements exceeded the modeled concentration where local sources were close to the measurement site. As yearly constant continuous emissions were injected in the model, actual emission pulses were not resolved. Furthermore, due to the coarse resolution of 200 km times 200 km in the Eulerian (grid box based) model ECHAM5 the plumes were diluted numerically in the direct surrounding of a source. Figure 3 is a good example for that deficit: The reprocessing plant Tokai is located only 60 km from the observation site Tsukuba. Thus the peaks of fresh plumes are not caught by the model. But the global background level is calculated perfectly as you can see from 1998 on, after the Plant at Tokai was shut down due to an accident. Figure 4 shows the comparison for weekly data at Miami. There is a good agreement, but in the 1980s the emissions in the model seem to be overestimated. By the way, it depends on the point of view what you want to evaluate: Either one can conclude from the good agreement that the model is correct, or you can see it as confirmation of the emission inventory. Of course both can be wrong as well with mistakes compensating for each other. The detailed results of all observation sites are documented in the project report [1]. In Addition, the results where compared to ship measurements of the 1980s and to few observations of vertical concentration profiles. Further publications in atmospheric sciences journals will follow.

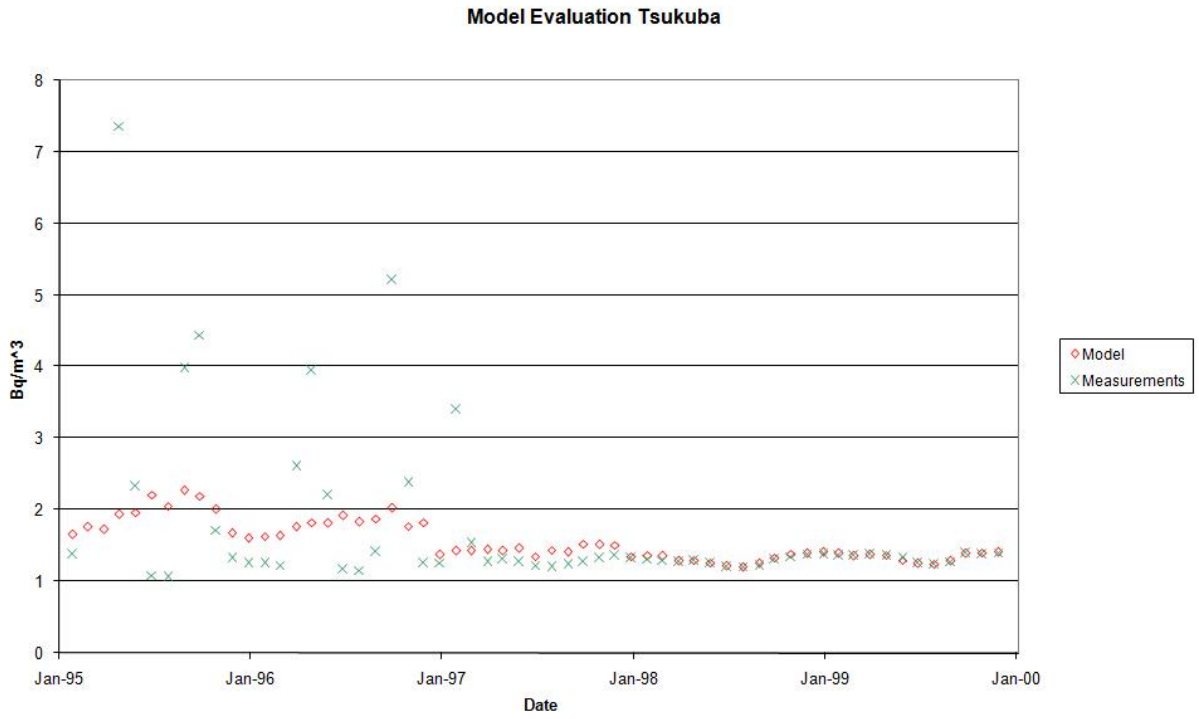

Figure 3 Comparison of observed (Measurements by BfS) and modeled monthly means at Tsukuba. The measured peaks are caused by the RPF Tokai at 60 km distance

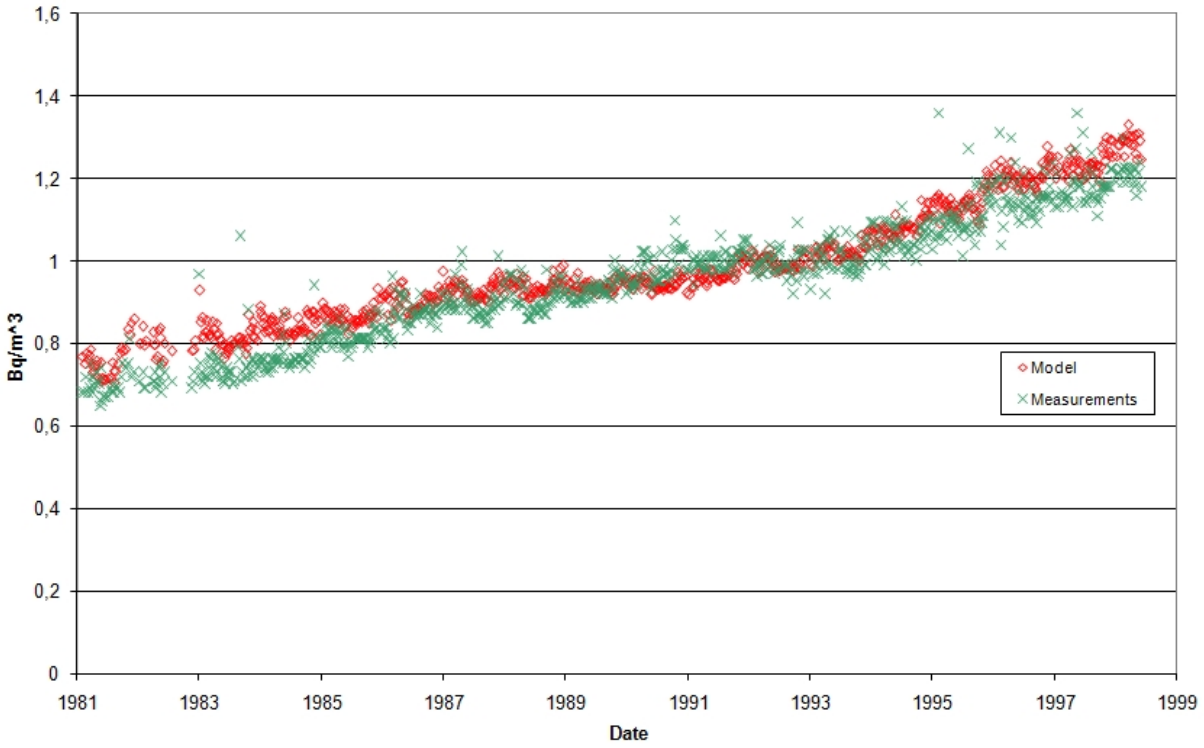

Figure 4 Weekly comparison of modeled and measured concentration at Miami, USA, Measurements by BfS

5. Detectability of additional sources

The key question of the first year of IAEA-project was to quantify detectability of small additional sources. For that, emission plumes were simulated for hypothetical emission pulses from arbitrarily specified source locations under various meteorological conditions. The applied model was the Lagrangian Particle Dispersion Model HYSPLIT, which is freely available to the scientific community [9]. In the evaluation of the ensemble of model runs the minimum detectable release was analysed in the maximum of the plumes. This is the activity of ^{85}Kr which has to be emitted to cause in the maximum of the plume a signal higher than three times the standard deviation of the background concentration. The variance of the background was calculated from the ECHAM5 data. The results on the minimum detectable release were compared with the following reference scenarios:
The specific source term under consideration was 20 TBq ^{85}Kr per kg separated plutonium, which is in the middle range of expectable source terms according to the diploma thesis of Paul Stanoszek [10].
An emission of 3.2 TBq corresponds to one dissolution campaign out of fifty to gain one significant quantity plutonium (8 kg) within one year, a release of 10 TBq for a slightly bigger plant and 100 TBq for a facility on industrial scale. The qualitative summary of the results is as follows: The big facility turned out to be detectable under nearly all conditions up to two days after the release, if the measurement is taken in the plume. For the scenario of the very small emission the detection situation is harder: The chances of detection become decrease significantly one day after emission. Thus, the inspectors have to be fast and go to the right places. Within the first two days after release the plume can travel several hundred kilometers. There are also the regional differences originating from different background situation. The detection situation is comfortable on the Southern Hemisphere (sometimes a plume stays detectable over more than 1000 km), and the situation is much more difficult in central Europe. To conclude, detectability can be assumed anyway within 50 km from a facility. The goal of applying advanced atmospheric transport modelling is to expand the area of potential successful detections to the range of 50 to 500 km. Details are presented in the project report [1].

6. Outlook

Comparison with measurements shows a good representation of background concentrations in the global model. Nevertheless detection of small additional sources remains challenging. A permanent measurement network with high density would still be too expensive (compare IAEA report STR 321 [11]). But there are different sampling scenarios possible: On the one hand, inspectors can try to catch plumes from suspicious sites; on the other hand they could take samples routinely where ever they go for inspections anyway and apply inverse modelling to identify possible source regions.
For both a more flexible measurement technology has to be developed. The beta counting applied so far needs quite large air samples. They can only be sized down with huge equipment in the field. A promising approach is the Atomic Trap Trace Analysis with this spectroscopic method it shall be possible to evaluate bottled 1-liter air samples in future. This method is also currently being developed at the Centre for Science and Peace Research at the University of Hamburg. The second year of the IAEA project will contain more detailed regional atmospheric studies assessing the localizability of additional ^{85}Kr releases and investigate the effectiveness of different sampling procedures.

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Automated processing of radionuclide data from the CTBT IMS network at the Finnish National Data Centre

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Abstract:

The Comprehensive Nuclear Test-Ban Treaty Organisation (CTBTO) is finalising its International Monitoring System (IMS) network. The purpose of the network is to detect any nuclear explosion, and thus give confidence to the effectiveness of the Comprehensive Nuclear-Test-Ban Treaty (CTBT). All data from the IMS is transferred to CTBTO headquarters, the Preparatory Technical Secretariat (PTS) in Vienna, where the data is analysed at the International Data Centre (IDC) and relayed to all interested state parties.

The radionuclide part of the IMS consists of 80 air filter stations, of which 40 are also equipped with noble gas (Xe) systems. The Finnish National Data Centre (FiNDC) receives all IMS radionuclide raw data (spectrums) from the IDC and analyses them in an automated analysis pipeline. The pipeline consists of the COTS software Unisampo and Shaman for air filter spectrums and the PTS developed Aatami software and in-house software for xenon data. Data and results are stored in a Linssi database, which is developed by STUK with affiliates and freely available. The pipeline automatically generates warning messages, whenever a suspect result is found during processing.

With the fully established network, the IMS air-filter stations will generate 80 final and approximately 700 preliminary HPGe spectrums per day. The xenon systems will add between 40 and 120 final HPGe and 3D beta-gamma coincidence spectrums daily, depending on the mode that the network will operate in. The FiNDC pipeline is automatically processing the data from the current network (approximately 80% complete) and refining the results to a level where the whole system can be effectively overseen by one human operator. This requires good tools for browsing the automatic analysis results and high confidence in the ability of the processing software to correctly analyse the data and detect any suspect information in it. This confidence has been built from several years of experience in running the pipeline for the air filter spectrums. The part of the pipeline processing xenon data is still under evaluation.

Keywords: CTBT; Gammaspectrometry; Air filter; Radioxenon

1. Introduction

The prospects for the ratification of the Comprehensive Nuclear Test-Ban Treaty (CTBT) are currently quite interesting. The USA are clearly stepping up their efforts towards ratification and several political statements have been made with a bright view of the possibility of the treaty entering into force (EIF) within a few years. The EIF of the treaty requires that all of 44 states named in the treaty ratify it [1]. Effective verification of treaty adherence is a required step in the process to gain the momentum needed for ratification in the nine countries that still have not done so.

The International Monitoring System (IMS) is built with the sole purpose of CTBT verification. Its major part consists of 337 monitoring stations, which are distributed evenly around the globe [2]. The design goal for the IMS was the ability to detect a 1 kT nuclear explosion, anywhere on earth [3]. The IMS utilizes four different monitoring techniques, Radionuclide, Seismic, Hydroacoustic and Infrasound. The fully built radionuclide part of the IMS consists of 80 radionuclide stations, which are all equipped with equipment for sampling and analysis of air-filters. Of these stations 40 also have equipment for

analysing radioactive xenon isotopes in the air. Additionally, there are 16 certified laboratories that can reanalyse interesting filter samples from the stations and perform system Quality Control (QC) functions. One of these laboratories (Fil07) is colocated with the Finnish National Data Centre (FiNDC) at STUK. In May 2009 about 80% of the full network and its radionuclide stations were operational (either certified or in test operations). 20 xenon systems are deployed and operational. [4]

All data from the monitoring stations is gathered in near real time at the International Data Centre (IDC) in Vienna, and analysed. All raw data and IDC analysis results are available to the signatory states of the Treaty, through their National Data Centres (NDC). It is up the National Authorities to decide what data they retrieve and how they use it. The FiNDC has specialized in analysing raw data from the radionuclide part of the IMS. For seismic expertise the FiNDC can get assistance from the Institute of Seismology at the University of Helsinki, who is also running the seismic IMS station PS17 in Lahti, Finland.

2. Operation of the IMS radionuclide network

Every sample at an IMS particulate station runs in a 72 hour cycle. First air is sampled for 24 hours, and then the air sample is stored in a waiting position for 24 hours, during which most of the natural radioactive materials (radon progeny) in the sample decay. During the last phase the sample is measured on a HPGe spectrometer at the sampling station. During the measurement, preliminary results are produced every 2 hours, which gives a total number of 12 spectrum files per sample, including 11 preliminary and 1 full spectrum. The stations are running continuously, so that sampling and measurement run a minimum of 22 h per day 365 days a year, excluding brakes for scheduled maintenance or equipment failure. [5]

3. Data retrieval

The data from the IMS monitoring stations is collected by the IDC through a Global Communications Infrastructure (GCI), built on small aperture satellite technology (VSAT). All raw radionuclide data (spectrums and status messages) is relayed by email from the IDC to FiNDC and other interested NDCs immediately on arrival at the IDC. NDCs can also request specific data from the IDC using different request methods. The data transfer from the IDC to NDCs goes either via VSAT or through internet based VPNs.

The IDC automatically analyses the final spectrums and releases an automatic radionuclide report (ARR), usually within 15 minutes from retrieval of the data. The IDC also performs manual analyses of these spectrums and releases a reviewed report (RRR), usually within 2 days, but these reports are occasionally significantly delayed. For spectrums that include abnormal amounts of anthropogenic radioactivity, the IDC also releases a more detailed report, called a standard screened radionuclide event bulletin (SSREB). The data flow, with time lags, the number of messages and the amount of data is shown in table 1. In addition to this, the IMS spectrums, IDC results and auxiliary performance data are available to NDCs on the IDC secure webpage.

4. Analysis pipeline at the FiNDC

All spectrums received at the FiNDC are immediately automatically analysed with the Unisampo and Shaman software (USS) [6,7]. Unisampo is a general purpose software for gammaspectrometry analysis and Shaman is an expert system for radionuclide identification. Together they provide a versatile tool for automatic and manual analysis of HPGe spectrums. USS has been adjusted especially for analysis of air filter samples and it has been tested and used extensively by several organisations, including the preliminary IDC at Arlington [8], the FiNDC, the Canadian NDC, and the Finnish national air monitoring program, run by STUK. Unisampo provides a hypothesis testing functionality, which is used by an alarm script, for generating automatic alarms when highly relevant nuclides are seen in a spectrum. Hypothesis testing is performed for all spectrums, including the preliminary ones, but the significance level for generating alarms is set higher for preliminary spectrums, to avoid high numbers of false positives, due to low statistics in the short preliminary

measurements. Alarms are divided into “alert”, “alarm” and “emergency” messages, which can be distributed to different mailing lists or mobile phone messages, according to the severity of the alarm.

	Number	Data	Ave.	Min	Max
		[GB]	time†	time†	time†
Number of IMS stations sending messages	52				
Total number of messages received at FiNDC	50608	0.5*			
-Spectrum messages	18098	2**			
-FULL spectrum messages (≈ number of samples)	1412				
-QC spectrum messages	1388				
-Preliminary spectrum messages	15298				
IDC reports	2833				
-ARR (automatic radionuclide reports)	1395		7 m	-51 m	67 m
-RRR (reviewed radionuclide reports)	1367		5 d	1 d	20 d
-SSREB (Standard Screened R.nucl. Event Bulletins)	71				
All processed data after cleanup		5*			
Al processed data before cleanup		19**			

Table 1: The number of messages from IMS particulate stations forwarded from the IDC to FiNDC. Data amounts and delay times for IDC reports, calculated from receipt of the raw spectrum. Storage requirements at FiNDC. All data are for January 2009, a typical month, except that the delay for RRRs is unusually large, because of backlog in IDC processing created during PTS December holidays.
† Time from when the FiNDC automated processing of the raw spectrum is finished, to receipt and storage of the IDC report in the file system at FiNDC.
* Compressed or partially compressed
** Uncompressed ascii

For some nuclides, there is a special treatment included in the alarming script, which is designed to minimize false or uninteresting alarms due to interfering natural nuclides, background ^{137}Cs etc. The number of spectrums analysed and the different alarms generated, during January 2009 (a typical month), are shown in table 1. The list of nuclides and energies used in the hypothesis testing is shown in table 2. Most alarms are false positives, either type 1 errors or due to some malfunction at the stations or inconsistencies in the dataflow. An approximate breakdown of the reasons for alarm messages in January 2009 is outlined in table 3.

Nuclide	Energy (keV)	Nuclide	Energy (keV)
Np-239	106.0	Cs-137	661.7
Ce-144	133.4	Mo-99	739.5
Tc-99m	140.4	Zr-95	756.8
Ce-141	145.4	Nb-95	765.8
Ce-143	293.4	Eu-154	1004.7
I-131	364.6	Eu-152	1112.0
Ru-103	497.2	Co-60	1173.2,1332.5
I-133	530.0	La-140	1596.0
Ba-140	537.4	Y-88	1836.2

Table 2: Nuclides and energies generating alarms in the alarm-test script at FiNDC

All analysis results, including results for preliminary spectra, are stored in the computer file system, for a limited time period. All final results, together with all relevant analysis data are stored in a Linssi database [9]. The Linssi database is a special purpose database for storing spectral data that has been developed in collaboration between STUK, Helsinki University of Technology (TKK) and Health Canada (HC). The database structure allows for storing of very detailed data relevant to the analysis.

Efficient use of the database requires tools and scripts for fast and efficient manual review of results from the automatic analysis and for extracting summary data. Such tools have been developed by the Linssi developers [10].

Number of spectrums in alarm-test = 16710	Emergency	Alarm	Alert
Total number of events / events from full spectrums	1/1	78/30	343/77
Check sources visible in measurement	1	46	-
Possible anthropogenic (mainly ^{99}Tc or ^{131}I)	-	10*	100
Type 1 or unresolved**	-	22	243

Table 3. Number of alarms generated by the alarm-script at FiNDC during January 2009. The alarms are divided into groups according to their severity. As a rule, detection of one single nuclide will generate an alert, whereas detection of two or five nuclides will generate alarms or emergencies accordingly. This list includes all generated alarms, both from preliminary and full spectrums
* The group “Possible anthropogenic” includes alarms where one nuclide is probably real, and the other probably a Type 1 error.
** Spectrums with lower than normal resolution often generate alerts, due to neighbouring peaks disturbing the hypothesis test. These are included in these numbers.

5. Manual review of analysis results

The FiNDC regular staff consists of only one person, who manages everything from spectrum analysis to technical and policy development and international cooperation in CTBT related matters. In addition to this there are a few people from the Nuclear Materials department and from the CTBT certificated laboratory Fil07 (also run by STUK), who can perform as stand-ins during the FiNDC managers travel or holidays, or perform specific tasks.

With these very limited resources it is not possible to do extensive manual analysis of all incoming spectrums, and neither is this necessary. The manual analysis that the IDC performs on every spectrum fills this basic requirement. The interest of the FiNDC is to gain confidence in the quality of the IMS and IDC radionuclide system and in our own capability to make an informed and speedy decision on the relevancy of any analysis results, in a situation where such a decision is needed. Therefore manual review is limited to spectrums where the automated analysis results indicate something out of the ordinary. Such an indication can be an alarm from the alarm script or the analyst can find something interesting when browsing through the newest results, a daily routine at FiNDC that does not take more than fifteen minutes, if nothing out of the ordinary is found. In most cases the manual review can be performed with the data browsing tools included in Linssi, but for in-depth review the actual analysis programs are used in manual mode.

The need for manual review can also rise from other causes. A political situation where a nuclear test is highly possible (e.g. the DPRK case in 2006) or a suspected seismic event will lead to a need to analyse spectrums from a certain region more carefully than normally, for a limited period of time. In such cases resources are allocated for performing in-depth manual review of normal spectrums from the geographic region of interest. Such cases are also very useful as motivating training opportunities for the FiNDC staff and stand-ins.

6. Current and future developments

The automated noble gas (xenon) measurement technology needed for the IMS network has been developed during the last ten years within the international Inge collaboration [11]. In addition to collection and measurement techniques also new analysis tools have been developed and are still under development. FiNDC is currently running the Aatami software developed at the PTS for the analysis of HPGe xenon spectra and an in-house developed code (bgpeaks) for the analysis of 3D beta-gamma xenon spectra. Both software can be used in either automatic or manual mode. They have been set up in automatic mode as pipelines with the Linssi database - the functionality resembles the airfilter analysis pipeline. However, this software has not yet undergone such extensive testing and adjustments that has made it possible for us to get constantly reliable results from the

airfilter pipeline. Moreover, we lack several tools needed to review these results as efficiently as those from the airfilter pipeline. We therefore still regard our xenon analysis capability as preliminary.

Atmospheric transfer modelling (ATM) is an important tool that gives possibilities to relate radionuclide measurements to possible source regions. ATM is of even higher importance with xenon measurements, because of the comparatively high prevalence of radioactive xenon in the atmosphere of the northern hemisphere. In cooperation with the WMO and leading weather centres, the PTS performs the calculations necessary to evaluate where the air particles collected in any measurement have originated. The PTS is also developing a tool, Webgrape, that allows NDCs to utilize these calculations efficiently. FiNDC has been using this tool for more than a year. The Webgrape tool is not used during normal operations, but only in situations requiring in-depth manual review. The next version of Webgrape will include preliminary data-fusion capabilities e.g. the possibility to show radionuclide station specific ATM results together with error ellipses from IDC analysis of seismic events.

An additional report format for the distribution of IDC analysis results is currently being developed and tested by the IDC and some NDCs. The format is based on extended markup language (XML) and the goal is to deliver sufficiently detailed information about the IDC analysis process to facilitate detailed interpretation at the NDCs. The XML format is easily computer readable and the FiNDC is currently developing a tool for converting the IDC generated XML into the Linssi format.

7. Conclusions

The FiNDC is an example of that it is possible to run all the processes required to efficiently follow and evaluate the IMS radionuclide network, and to uphold sufficient capabilities to make informed decisions in case of a real CTBT relevant event, with minimal resources. All automated and interactive processing required can be run in one modern Linux based computer server. When the degree of automation is high, daily routine work with the data can be minimized to less than an hour, even including fast browsing through all relevant results. Radioxenon measurements and ATM are even more important parts of the IMS than initially believed, because radioxenon escapes also from underground tests and can provide the smoking gun evidence of testing, which is difficult to achieve with other methods. Therefore the ongoing work to develop tools for fully utilizing these results from the IMS is of high importance.

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SESSION 15

ILLICIT TRAFFICKING AND PORTABLE DETECTION SYSTEMS

JRC support to EC policy in the field of nuclear and radiological security

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Abstract

A policy package on combating CBRN terrorism is being developed through the CBRN Task Force, which brings together experts from the authorities of the EU Member States, the private sector and the EU institutions. The Commission intends to submit it for adoption in June 2009. The radiological/nuclear side features are prominently in these efforts with the objective to limit the risk of malicious acts, including terrorism, involving radiological/nuclear materials. The Action Plan on radiological/nuclear risk reduction is being developed by the RN sub-group of the CBRN Task Force. The work of the RN group is divided into three broad thematic areas such as Prevention, Detection and Response. Under each of these thematic areas, a number of specific issues have been addressed. The JRC will play an important role in the implementation of these policies: i) Testing of the EU equipment used for the detection and identification of nuclear material will be carried out in collaboration with IAEA by the JRC IPSC Ispra. This project will open the way to an EU certification scheme of the detection equipment. ii) The JRC Institute for Transuranium Element already possesses an advanced nuclear forensic capacity, which will be made available to Member State authorities. ITU should then play a central role in a network of research and forensic institutes for the purpose of performing measurements/analysis on RN and advising on forensics. iii) Both ITU and IPSC are providing training on nuclear security since decades. They will constitute the two pillars of the future EU security training centre that should be established by the end of this year 2009. These activities are complemented by new R&D projects funded by the JRC Euratom FP7 and are in accordance with the new R&D recommendations made by the European Security Research and Innovative Forum (ESRIF).

Furthermore, the international dimension of nuclear security has brought the JRC to enhance its collaboration with the major actors such as IAEA, and authorities in the US institutions and the Russian Federation. This collaboration has and will contribute to the successful implementation by JRC of many projects emanate from the various instruments the EU has set up such the Instrument for Stability, which provide the EU with funds and mechanisms to address global and trans-regional threats.

This paper highlights the role of JRC in implementing some of the new recommendations on the security policy of RN materials made by the EC and EU Member States in the CBRN Task Force. The paper underlines the important role of internationally coordinated projects related to RN detection and emphasises the technical and scientific role of the JRC in support to EU external security policies.

1. Introduction:

The Commission intends to submit a policy package on combating CBRN terrorism in June 2009. The package will consist of: General/horizontal communication on CBRN as well as an Action plan on Chemical, Biological, Radiological and Nuclear risk reduction
The policy package has been developed through the CBRN Task Force, which brings together experts from the authorities of the EU Member States, the private sector and the EU institutions. The active participation of the private sector is encouraged in the spirit of the Public-Private Security Dialogue.
The part of the Action Plan on radiological/nuclear risk reduction has been developed by the RN sub-group of the CBRN Task Force. The work of the RN group has been divided into three broad thematic areas such as Prevention, Detection and Response. Under each of these thematic areas, a number of specific issues have been addressed. For the prevention the areas covered are related to the Control over sources, Exchange of information, Physical protection of sources. Among the detection topics for the detection technologies, enhancing awareness among first line officers on what to look for when searching for radioactive/nuclear sources, Exchanging information on processes and procedures, certification, testing and trialling schemes; and standardisation. Areas of concern for Preparedness and Response are mainly those related to National response plans, communication among the various organizations and actors involved; exchange of information and nuclear forensics. Research priorities and trainings are horizontal issues and are part of the recommendations made for the three thematic areas.
The JRC is closely associated to the related CBRN policy (for its RN part). JRC has carried out a Radiological Vulnerability Risk Assessment Study in EU 27, which supported the work within the CBRN Task Force. Moreover, in support to EU MS in the field R&N, the JRC has been entrusted by the EC (DG-JLS Justice Freedom and Security) and Member States for the establishment of a European Radiological/Nuclear Security Training Programme for the Law Enforcement Community – EUSECTRA. The objective of the action is to establish a security training programme applicable to the law enforcement community. The second important JRC policy support project concern the evaluation and comparison of the performance of available detection equipment relevant to nuclear security – the ITRAP+10 project. The JRC is very well placed for undertaking this project as it has access to the necessary radioactive/nuclear materials needed to perform the tests and it can ensure that no conflict of interest takes place in the testing of equipment manufactured in various states.

The EU is concerned by the risks that not only states but also non-state actors may be misusing peaceful nuclear technologies to develop nuclear weapons or may use radioactive substances for making ‘dirty bombs’ with high potentially damaging effects.
In the context of the EU *Common Foreign and Security Policy* (CFSP), the EU has adopted four Joint Actions in support of the IAEA nuclear security fund totalling a financial commitment of approximately 23 million EUR [1]. These contributions have made the EU the major donor to this IAEA programme. The geographic focus of these activities has so far been on countries in Africa, South-East Europe, Caucasus, Central Asia and the Mediterranean, and is now expanding to countries in South-East Asia. The EU continue also to co-operate with the UNSC 1540 Committee and the UN Office for Disarmament Affairs in organising regional and sub-regional seminars in order to promote awareness of the UNSC Resolution 1540 requirements [2] and in assisting countries to comply with them.

In the context of the *Instrument for Stability* (IfS) [3], which supports CFSP objectives, as well the European Security Strategy [4], the European Commission has engaged in comprehensive assistance activities, which are, inter alia, aimed at addressing the risks presented by nuclear terrorism. Currently, projects for the redirection of WMD scientists in the CIS, development of capacity to combat illicit trafficking in CBRN materials, and for strengthening of export controls globally are underway. The geographic scope of these projects is being enlarged beyond the CIS and complemented, inter alia, by the establishment of CBRN safety and security training centres in South East Asia and Mediterranean countries.
The JRC is supporting the implementation of projects under the IfS on combating the illicit trafficking of nuclear and radioactive materials in the CIS countries as it was before under the TACIS programme and similar projects are being dedicated to Caucasus, Central Asia and the Mediterranean countries. The projects concern the provision of equipments as well as related trainings.

In order to identify areas for action and to consult potential beneficiary countries, ensuring coherence between different actions under IfS priorities and other international programmes, the *Expert Support Facility* (ESF), was established under the first IfS Indicative Programme.

The JRC and experts from EU MS, together with Commission representatives visited few potential beneficiary countries and consulted them at political and expert levels in regions from Northern Africa, the Middle East, and South-East Asia. In addition, the JRC contributed to many expert workshops — on nuclear smuggling, export control and the redirection of weapons scientists — that were organised by the ESF. These workshops involved numerous experts not only from the EU and potential beneficiary countries, but from important stakeholders such as the United States, Russia and relevant international organisations (e.g. the IAEA and WCO).

2. The JRC support to intra EU policy:

2.1 European Security Training Centre (EUSECTRA)

The JRC, in line with its mission to provide technical scientific support to the EU policies and also with its obligations mentioned in the Euratom treaty, has a dedicated strategy in the field of education and training. Yet, the JRC has a long standing tradition in providing training in nuclear safeguards and security in support to EURATOM and IAEA inspectors, as well as to law enforcement bodies of the New Member States during the Enlargement process and still now. It has also designed the Russian Methological and Training Centre and contributed to its establishment.
The importance of training at all levels for both EU internal and external nuclear security has conducted the JRC to offer its establishments by the creation of a European training centre which will be dedicated to nuclear security issues of EU27 and countries that benefit from the EU support under the instrument for pre-accession and the instrument of stability.
The overall objective of the proposed training centre is to raise awareness among the concerned community and *in fine* to build a security culture on the model of the safety culture that was established and shared between the nuclear operators following major nuclear accidents like Three Miles Island The EU Security Training Centre will operate at the two JRC sites, JRC-IPSC (Ispra, Italy) and JRC-ITU (Karlsruhe, Germany) in order to benefit from the specific expertise and from the infrastructure available at the respective sites. This complementarity will assure a comprehensive coverage of the field of nuclear security. The training courses will be offered to the EU27 member states as well as to beneficiary countries of the TACIS and IfS (Instrument for Stability) support programs.
New infrastructures will be required and detailed courses be designed during the establishment process.

2.1.1 Training Programme:

a) Training related to the first line of defence
The first line of defence corresponds to the prevention of the diversion of nuclear materials. The usual corresponding measures are based on proliferation resistance, physical protection and traditional Safeguards. As already underlined, the JRC has a long standing experience in the latter. It is also involved in the two first areas through its responsibilities in the Generation IV International Forum. However, these areas remain mainly within the competence of EU Member States and thus their involvement will be essential to the success of the project. Cooperation with US-DOE would be also very appropriate, as DOE has already such training centres and has establish one in Obninsk (Russian Federation).

b) Training related to the second line of defence: the detection
The intention is primarily to develop hands-on training for front line officers and law enforcement services involved in the detection of and response to cases of illicit movement of nuclear and radioactive materials (NRM). This core course will be based on and developed with the support of the US DoE HAMMER Training Centre. A shorter specific course dedicated to them will be proposed accordingly. The sustainability of the support in the country of interest will be guaranteed by a higher level and specific session dedicated to the transfer of know-how (train-the-trainer concept). This transfer could also be envisaged as a dissemination of the training centre outside the European Union, in line with past efforts to establish and/or support regional training centres.
c) Training of Measurement Experts
Once nuclear material has been intercepted and the initial response by front line officers has been carried out, the support of a mobile expert support team (MEST) is required for further handling of the case. MEST team members should have measurement expertise using mobile equipment, they should also be aware of preservation of forensic evidence and they should initiate the further steps to be

taken. Target audiences for this training are measurement experts from health and safety institutions, from measurement laboratories, from radiation protection services, from environmental protection services or from research institutes. The specific topics to be covered are: radiological advice, preservation of evidence, categorization of the material, sample taking and assistance in the "crime scene management.

d) Training related to the response plan
The administrative infrastructure for detection and response needs to be described and the responsibilities of the difference authorities and services have to be defined. Based on the ITWG's Model Action Plan and on the corresponding IAEA documents, a generic national response Plan will be developed. Target audiences for this training are law enforcement, regulators, health and safety institutions and measurement laboratories. The specific topics to be covered are: the Model Action Plan, definition of roles and responsibilities, definition of communication lines, definition of processes, nuclear forensics awareness.

e) Training in nuclear forensics
Nuclear forensics comprises sample taking, sample analysis, interpretation and attribution. It is a key element in the response process, providing useful information for prosecution and for enhancing prevention at the place of diversion of material. Target audiences for this training are experts from nuclear research institutes and from nuclear measurement laboratories. The specific topics to be covered are: chemical and radiochemical analysis, particle analysis, trace and ultra-trace analysis, material properties, fuel cycle knowledge, data interpretation, source attribution, drafting of expert opinion.

2.2 ITRAP + 10

2.2.1 Goal of ITRAP+10:

The European Commission in particular through its Joint Research Centre (JRC) is supporting the fight against illicit trafficking of nuclear and radioactive materials and is implementing specific projects dedicated to border monitoring. The European Commission attempts to put forward ideas on how to improve access of law enforcement authorities to high quality detection tools. Testing and qualifying the related equipment remain crucial for the credibility and the usefulness of the technology in the field of combating the illicit trafficking of Nuclear and Radioactive materials.
The EC ITRAP+10 project will undertake an evaluation and comparison of the performance of available detection equipment relevant to nuclear security. The results will provide an independent assessment of the available detection equipment on the market which will serve as a reference for regulatory and other Member State authorities to identify equipment to address their particular needs, and help to introduce common standards at a European level. In parallel, the manufacturers of detection equipment will receive feedback and recommendations to improve performance and sustainability of the equipment. This will strengthen European industry by making their products more competitive. This action will provide a much needed update to the ITRAP project undertaken 10 years ago to account for changes in technology.

2.2.2 Equipment to be tested

The EU27 companies with series productions of the selected type of equipment will be invited to participate to the ITRAP + 10. Participation in the tests will be free of charge, financed by EU for European companies and probably by US DOE and DHS for US manufacturers. FSU countries might be financed through TACIS or its follow up programs. ITRAP+10 will start in 2009 and might take several years.
Flexibility will be given between the different types of instrument to optimize the use of the available budget.
The selection of equipment and the decision on the final number per type will be done during a dedicated coordination meeting.

2.3 Nuclear Forensics

Nuclear forensics is a methodology that reveals information inherent to nuclear material. Nuclear material has either been subject to technological processing or is entirely of anthropogenic origin.

Consequently, nuclear material carries "toolmarks" or "fingerprints" of the process it was subjected to. Uranium fuels are examples of the first category, while plutonium belongs to the second category. Both elements contain fissile isotopes, substantiating the broad interest in the history, origin and intended use of these materials. Historically, the first area of application was related to military intelligence. Today, however, a much broader range of applications is conceivable:
In non-proliferation issues, nuclear forensic methodologies are applied. The investigation of particles of highly enriched uranium found in Iran provided hints on Iran's clandestine nuclear programme. The measurement of chemical impurities is increasingly applied in nuclear safeguards. Particularly, samples of uranium are analysed in order to establish relations between different samples and check consistency of the impurity pattern with the declared processes.
Also environmental samples can be subject to nuclear forensic investigations. As demonstrated by JRC, particles found on the seabed and occasionally on the beaches of Dounreay could be attributed to the Material Testing Reactor at Dounreay using electron microscopy in combination with elemental analysis (by X-ray fluorescence).
The Nuclear Security area is certainly the most prominent application of nuclear forensic methods. This comprises illicit trafficking but includes also (prevention of) nuclear terrorism.
The JRC with its Institute for Transuranium Elements has been involved in nuclear forensic investigations for almost two decades and provided support to member state authorities in more than 30 cases. Nuclear forensic investigations have to be considered as part of a comprehensive set of measures for detection, interception, categorization and characterization of illicitly trafficked nuclear material. As mentioned above, nuclear forensic analysis may result in important conclusions on the origin of the material and thus provide the most essential contribution to the prevention of future diversions from the same source. The JRC has provided support to member states in through investigation of seized nuclear material. A number of joint analysis agreements were concluded (in line with the IAEA's recommendation as expressed in the document Nuclear Security Series No.2) in particular with the new EU member states. Moreover, training in development of response plans, in nuclear forensics awareness and in technical nuclear forensics is being provided to experts from member states and from other states upon request. The further development of nuclear forensic capabilities is fostered by close collaboration with law enforcement services on the one side and by networking and scientific exchange with other nuclear forensics laboratories. To this end, the JRC has concluded (and is concluding) a number of collaboration agreements with national and international expert organisations. JRC is also co-chairing the nuclear smuggling International Technical Working Group (ITWG), which is the main international forum for nuclear forensics experts.

3. Support to the implementation of the EU foreign policy

3.1 Combating illicit trafficking of RN

Nuclear security has been included in the TACIS (Technical Assistance to the Commonwealth of Independent States) program in 1999 with a pilot project on counteracting non-authorized transfer of nuclear materials in Ukraine. The new 2005 TACIS project developed further the EU support in the field with the implementation of specific projects dedicated to the fight against illicit trafficking of nuclear and radioactive materials and border monitoring activities in Azerbaijan, Armenia, Belarus, Georgia, Moldova, Russian Federation and Ukraine.
The following selected activities illustrate the EU support under this TACIS 2005 project (non exhaustive list):

- Three Ukrainian border crossing points, jointly selected to complement the US support program to Ukraine, will be equipped with stationary detection systems (Radiation portal monitors for vehicles and pedestrians) and associated hand-held equipment for secondary inspection. Front line officers will be trained accordingly.
- Necessary detection equipment will be deployed at Yerevan International airport and corresponding training provided.
- The JRC will jointly provide support to Georgia with the US Second Line of Defence program at agreed selected location.
- Support to national expert laboratories is foreseen in the Russian Federation and Ukraine.
- Equipment for border crossing points and provision of mobile laboratories are planned in the Republic of Belarus.

With the end of the TACIS program in 2006, the new *Instrument for Stability* allowed the finalisation of the on-going projects and the extension of the EU efforts to other identified geographical areas. The new activities are focussing on the Mediterranean Basin where border monitoring activities will be deployed this year in two selected countries. In parallel, fact-finding missions will be conducted in the ASEAN region (South East Asia) to draft an Action Plan in close coordination with other major international donors.

3.2 Expert Support Facility

The overall objective of the *Expert Support Facility* (ESF) is to contribute to the objectives of the long-term component of the *Instrument for Stability*, such as 'Counteracting Global and Trans-regional Threats' and 'Proliferation of Weapons of Mass Destruction' of the IfS'.
The ESF ensures the availability of a pool of expert organisations, readily deployable for needs assessment and fact-finding missions, as well as execution of feasibility studies, allowing interaction with beneficiaries and coordination with other donors. The tasks of the ESF are carried out by EU Member States specialised public bodies and the JRC, which makes available experts from most of its institutes.
Since the beginning of ESF activities in early 2008, the contribution of the Joint Research Centre (JRC) has been continued, in support to activities in the following areas: nuclear security threats, illicit CBRN trafficking and proliferation, CBRN materials security and Critical maritime routes.
The JRC has in particular provided support for a number of specialised workshops in the areas of Nuclear Smuggling and Illicit Financing, Export control and Redirection of weapon related scientists.
JRC-IPSC nuclear security experts have taken part to some fact-finding missions in North Africa, Middle East and South-East Asia, which served to provide background information for the definition of the new IfS Indicative Programme 2009-2011, and of the projects launched in 2009.

4. Conclusion:

The JRC, in accordance to its mission to provide technical and scientific support to the EU policies, is closely associated to the related CBRN policy (for its RN part). In support to EU MS in the field of detection, the JRC is working on the ITRAP+10 project in which world-wide equipment will be tested for their performances and limits. The JRC is currently elaborating a study for "enhancing member states capabilities on detection and response to nuclear illicit trafficking and this includes nuclear forensics. As the training on security matters is a key issue in the CBRN agenda, the JRC is going to establish a European Security Training Centre which will focus at its starting phase on nuclear and radiological security.
Outside the EU, various instruments have been established by the European Commission, such as the *Instrument for Stability* which provides funds and mechanisms to address global and trans-regional threats. the JRC will continue to implement the EC projects related to nuclear safeguards and fight the illicit trafficking of nuclear and radioactive materials. The Instrument of Stability has no geographical limitation (in comparison to the TACIS program) new regions will be supported via many projects in the field of non proliferation of WMD and CBRN risk reductions.

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The Development of a Handheld Laser-Induced Breakdown Spectroscopy (LIBS) Device for Nuclear Security and Safeguards Applications

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Abstract

Laser-Induced Breakdown Spectroscopy (LIBS) uses optical emission spectroscopy of the induced plasma to identify elements. It is an analytical method used increasingly by research groups and industry alike, offering many advantages: no sample preparation; small and rapid sample taking; real-time results; in-situ and stand-off analysis; and minimum operator training. Because of these traits, the Canadian Safeguards Support Program (CSSP) recognized that International Atomic Energy Agency (IAEA) inspectors and Member States' security staff could use LIBS technology in performing their duties.

Laboratory results have shown that LIBS technology can identify indicators and signatures of nuclear clandestine activities, even with a moderate-resolution spectrometer. These findings, as well as the quick identification of triuranium octoxide (U_3O_8 or "yellowcake") powders by using pattern-recognition chemometric procedures, are reported within this paper. The paper also presents the progression in developing a handheld instrument for field operations to be used by inspectors, nuclear security personnel and border crossing staff.

Keywords: Laser-Induced Breakdown Spectroscopy; nuclear clandestine activity

Introduction

Terrorist and proliferation activists are employing more sophisticated means[1] than those used in the past to achieve their objectives. Border Security Services, First Responders and Regulators need to adapt to this challenge and to seek technologies that can provide quick and accurate information, in order to prevent clandestine activities or initiate rapid responses to them. As a nuclear regulator, the IAEA recognized this need and requested its Member States to assist it with finding novel technologies to perform detection and inspection duties.[2] The CSSP accepted this challenge, introducing the IAEA to a technology outside typical radiation detection methods: Laser-Induced Breakdown Spectroscopy (LIBS).

With the invention of lasers in the early sixties, scientists have been exploring the concept of element identification by way of the microplasmas produced via the LIBS technique. However, this LIBS research really began progressing during the last decade due to technology miniaturization, and it is now being adopted by many research groups in various scientific fields and laboratories around the world.

Technique

In simple terms, the LIBS technique employs a high-power pulsed laser focused on a sampling material to be analyzed (see Figure 1). A very small amount (measured in nanograms) of the material is vapourized, producing an ionized gas — the plasma. The light emitted by this plasma can then be analyzed by an optical spectrometer, identifying the elemental composition of the sample. In turn, this allows the material composition to be identified with the appropriate software.

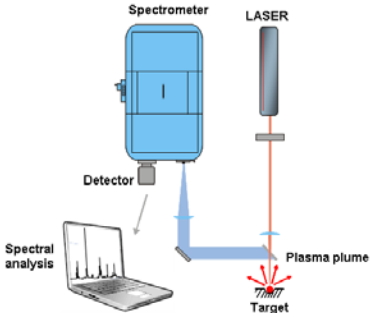

Figure 1: A LIBS Schematic

This LIBS technique has many advantages over other analytical methods:

- No sample preparation is required since it uses a high-powered laser.
- The laser beam is so finite that only a small amount of the material (typically a fraction of a nanogram or less) is actually sampled.
- The sampling material can be any form: gas, liquid, solid, conductive or non-conductive.
- The laser focusing provides spatial resolution, thereby allowing the study of elemental distributions on surfaces or depth-resolved concentration profiles within the material itself.
- Contact with the sample is not necessary, so analyses can be performed at safe distances in hostile environments.
- Samples do not have to be initially removed to another location, unless for confirmation purposes.
- The technique is simple and does not require extensive operational training.
- A single-shot analysis or multi-shot analysis can be performed within seconds.

Proof-of-Concept Testing

In seeing how these advantages could benefit the IAEA, the CSSP, in cooperation with Canada's National Research Council - Industrial Materials Institute (NRC-IMI) division, sought first to perform a "proof of concept" of the LIBS technique. Upon a successful proof of concept, the next goal was to develop a handheld LIBS instrument. The project initially examined the ability of LIBS to discriminate the nuclear clandestine indicator of maraging steel from that of other metals as well as other classes of steel.

Using an échelle spectrometer coupled to an intensified charge-coupled device (ICCD) camera with settings of 1-µs delay and 10-µs integration time, a 200-mJ Nd:YAG laser with MatLab 7.8 software produced typical LIBS spectra of the sampling material (see Figure 2).

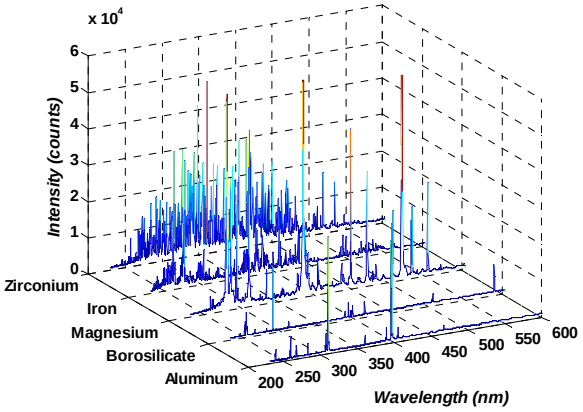

Figure 2: Typical LIBS spectra

A quick visual review of the spectra in Figure 2 allows one to easily distinguish the significant differences between these material samples. When more materials with similar structures (classes of steel) are sampled, this kind of identification can be quite difficult. However, the use of chemometrics methods can ease this tedious discrimination activity.[3] Using a Principal Components Analysis (PCA) method, each of the spectra was broken down to determine the respective principal components. A Soft Independent Modeling of Class Analogy (SIMCA) method[4] employing a multiple PCA modeling component[5] was used to sort the

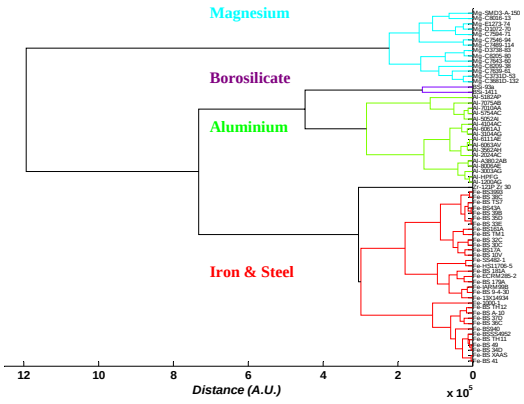

Figure 3: A Discriminant Analysis (SIMCA) of metal alloys

complex data. A first-level application of this method resulted in a separation of the raw data by using Mahalanobis distances with a centroid linkage (see Figure 3).

Although this level-1 run was able to sort most of the material, samples such as chromium were attributed to the unknown class. A level-2 SIMCA run was performed to resolve the primary class placements into sub-class assignments.

The results of this level-2 run resolved these misclassifications while further discriminating the samples. Since the aim was to determine if LIBS could distinguish maraging steel, only the results for the steel calculation are shown here (see Figure 4).

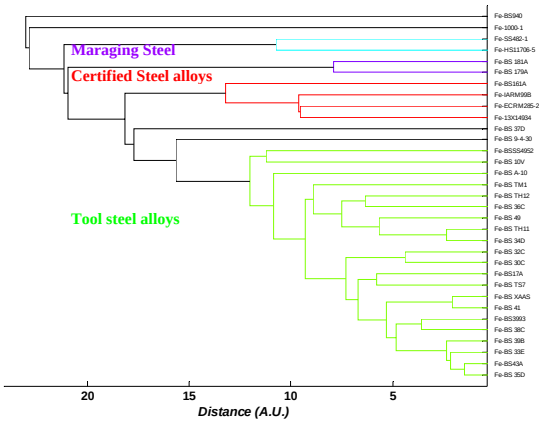

Figure 4: Discriminant Analysis (SIMCA) of steel alloys

The maraging related steels, Fe-BS 179A and Fe-BS 181A, are clustered together and depicted in purple in the dendrogram in Figure 4. Because of their unique chemical compositions, these steels were relatively easily sorted from the other iron/steel alloys. It should be noted that the SIMCA level-2 run was able to sort out the pure iron samples (Fe-1000-1, depicted in black), the certified reference material (Fe-BS 161A, Fe-IARM-B, Fe-ECRM 285-2 and Fe1x14193, depicted in red), and the other iron/steel alloys (depicted in green). This latter classification could be further resolved using this method, but such sorting was beyond the objective of this initial phase. Having successfully demonstrated that the LIBS system, using the appropriate analysis procedure, could discriminate between various materials within the same class, the project began to focus on the radioactive actinides of IAEA-designated nuclear indicators and signatures.

This second part of the proof-of-concept study required transporting the LIBS system to the IAEA Safeguards Analytical Laboratory (SAL) in Siebersdorf, Austria, in order to investigate actinides. The study went further to identify the ability of LIBS to determine the origins of various triuranium octoxide (U_3O_8 or “yellowcake”) powders. A slightly different physical configuration from the previous testing was used (see Figure 5).

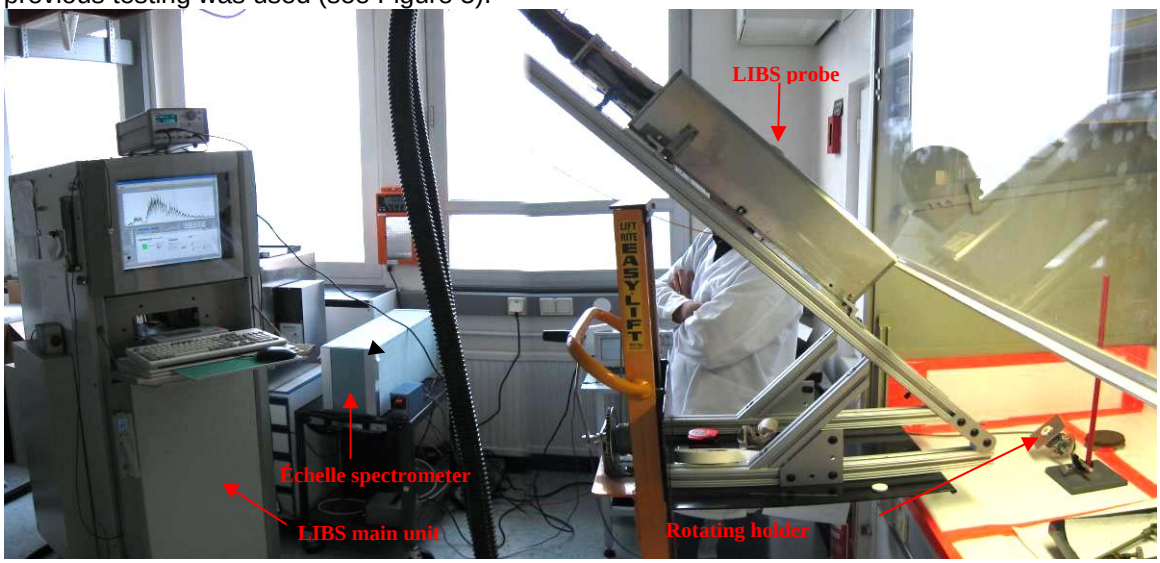

Figure 5: LIBS experimental setup at IAEA SAL

Two different spectrometers were used within this experiment: a Mechelle 5000 (Andor) spectrometer and a Czerny-Turner spectrometer. The Mechelle 5000 (Andor) spectrometer was equipped with an ICCD camera (Andor iStar DH-734) where the ICCD images were resolved using the IMI-LIBS 2007–Échelle software, with a 2-µs delay and a 30-µs integration time. The resulting spectra were composed of 22 063 pixels intensities between 230 and 870 nm. This high-resolution spectrometer was employed to
ensure the LIBS technique could distinguish the differences between the various yellowcake samples. Alternatively, the Czerny-Turner spectrometer, operating at a 20-µs integration and a 2-µs delay, was employed to represent the performance of a possible handheld field device. This spectrometer integrated the resulting light perpendicular to the sample surface, duplicating the same configuration to be employed for a handheld version. All samples were freshly prepared using a diluted 65% HNO_3 solution deposed on a ceramic sample holder and oven dried. Each LIBS analysis consisted of 100 laser shots creating a circle of 4 mm in diameter on the rotating sample holder.

Figure 6 depicts a typical set of U_3O_8 spectra using the Échelle spectrometer,. It is important to note that most of the spectra are composed of the more than 60 000 tabulated uranium lines. A quick scan of the spectra reveals possible differences in the amount of uranium as well as the presence of other elements within the various origins of yellowcake. Typically, yellowcake contains 70 to 90% U_3O_8 by weight[6] along with other various impurities such as uranyl hydroxide, uranyl sulphate, sodium para-uranate, uranyl peroxide and various uranium oxides. These impurities, along with pattern recognition analysis software, allow one to determine the yellowcake's origin.

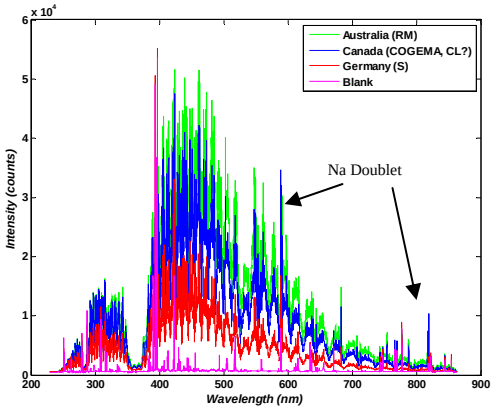

Figure 6: Sample of U_3O_8 spectra

Some PCA-based clustered techniques have limited ability to discriminate small variations among classes and within classes[7] (used in the first analysis of the data; therefore, a Partial Least Squares-Discriminant Analysis [PLS-DA] was chosen. The pattern recognition results obtained by the PLS-DA for 3 data sets (replicates) of each origin of yellowcake are plotted in Figure 7.

Figure 7 allows us to conclude that each data sample was correctly discriminated, with each having a probability higher than 0.75 of belonging to the right class, and a probability of less than 0.12 of belonging to the other classes. Thus, it was concluded that the LIBS technique could identify the origins of U_3O_8.

The PLS-DA model's robustness for each spectrum was also challenged. Table 1 depicts the mean probabilities for a single-shot spectrum, classified in percentages.

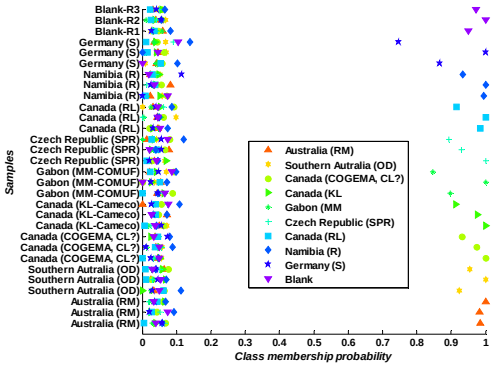

Figure 7: Origin identification of yellowcake

Mean Probability Sample	RM %	CL %	KL %	RL %	SPR %	MM %	S %	R %	OD %
Australia (RM)	**100**	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Canada (CL)	0.0	**98.4**	0.3	0.0	0.0	0.3	0.4	0.0	1.0
Canada (KL)	0.0	0.0	**97.6**	0.3	0.0	0.0	0.0	0.0	0.3
Canada (RL)	0.0	0.0	0.0	**99.7**	0.0	0.0	0.0	0.0	0.0
Czech Republic (SPR)	0.0	1.6	1.4	0.0	**100**	0.0	0.4	2.2	0.0
Gabon (MM)	0.0	0.0	0.3	0.0	0.0	**99.7**	0.0	0.0	0.0
Germany (S)	0.0	0.0	0.0	0.0	0.0	0.0	**98.9**	0.0	0.0
Namibia (R)	0.0	0.0	0.3	0.0	0.0	0.0	0.4	**97.8**	0.7
Southern Australia (OD)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	**98.0**

Table 1: Mean probabilities for single-shot spectrum classification

A greater than 97.6% mean probability of correct attribution was found for a single-shot PLS-DA, with less than 2.2% mean probabilities of false attributions. Based on this analysis and a literature search, it appears that a LIBS system using a combination of SIMCA and PLS-DA would constitute a powerful approach to pattern recognition, considering SIMCA's robustness and the detailed discrimination abilities of PLS-DA for small spectra variances.

To simulate a low resolution spectrometer (for field use) that could classify the origins of yellowcake, the binning was increased, so that 1 pixel represented approximately 22 pixels at the higher setting. Figure 8 compares the simulated-resolution spectra (blue line) to the higher-resolution spectra (green line), showing that distinctions can be made with a lower-resolution spectrometer.

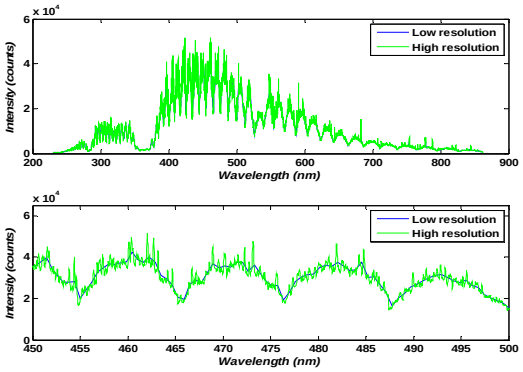

Figure 8: Mean LIBS Ranger Mine yellowcake spectra using high and simulated low resolution spectrometer settings

The results obtained using the Czerny-Turner spectrometer with an increased binning setting demonstrates the ability to predict the uranium enrichment of samples. The data for the different uranium enrichment samples was correlated to the isotope enrichment using Partial Least Square (PLS) multivariate regression model, where the different U_{235} enrichment spectra are depicted in Figure 9.

The accuracy obtained for the predicted amount of U_{235} is globally less than 20% relative, and becoming less than 5% relative for 20% and 93% enrichment. This analysis shows a moderate-resolution spectrometer LIBS system can reasonably predict the enrichment of an U sample using a PLS regression model, with an accuracy of better than 5% for a U_{235} amount greater than 20%, and with an accuracy of better than 20% for a lower U_{235} content.

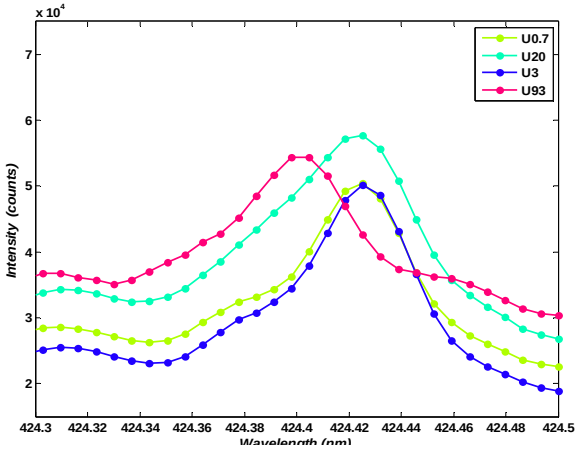

Figure 9: U enrichment LIBS Spectra

Using the high-resolution spectrometer, the project evaluated the limit of detection of the LIBS instrumentation for various actinides. Figure 10 presents the mean LIBS spectra for different thorium concentrations.

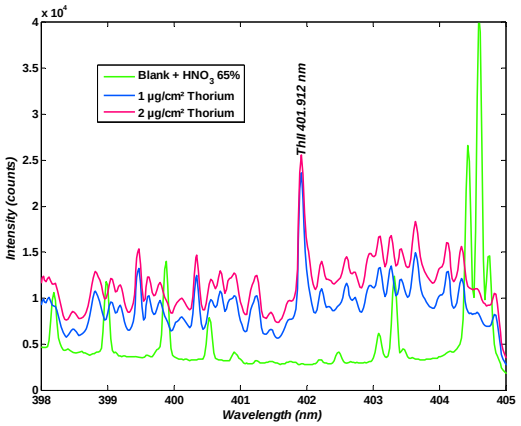

Figure 10: Th Concentrations LIBS Spectra

Figure 10 clearly demonstrates that the line for Th 401.912 mm is quite evident and matches the National Institute of Standards and Technology's database.[8] It should be noted that the difference between 1 and 2 µg/cm^2 is not significant. Although the LIBS system cannot distinguish the amount of the sample, it can identify thorium in quantities as low as 1 µg/cm^2. Table 2 shows the results for the remaining tested actinides.

Element	Detection Limit (ng/cm^2)
Uranium	100
Thorium	1000
Plutonium	25

Table 2: Actinide detection limits

Overall, this proof-of-concept study demonstrated that a moderate-resolution LIBS device, along with a multi-variant analysis method, can:

- distinguish various actinides at levels as low as 25 ng/cm^2
- disseminate specific samples from various classes and its own class
- determine the isotopic enrichment of uranium
- identify the origins of various yellowcake samples

Instrument Development

Due to this success with the lower resolution, the construction of a Handheld LIBS device (HLD) is now being undertaken. Figure 11 depicts a conceptual drawing of the instrument. The device will have sufficient safety mechanisms to allow operation without personal protection, but safety glasses will be provided as a precaution. The top five materials identified by the chemometrics software from its database will be displayed along with respective confidence levels. It is envisioned that the device will be light in weight and sufficiently robust to be used in the field. A HLD prototype will be constructed and tested in the summer 2009. Modifications will be proposed, and the first production version of the HLD is expected to be donated to the IAEA in December 2009.

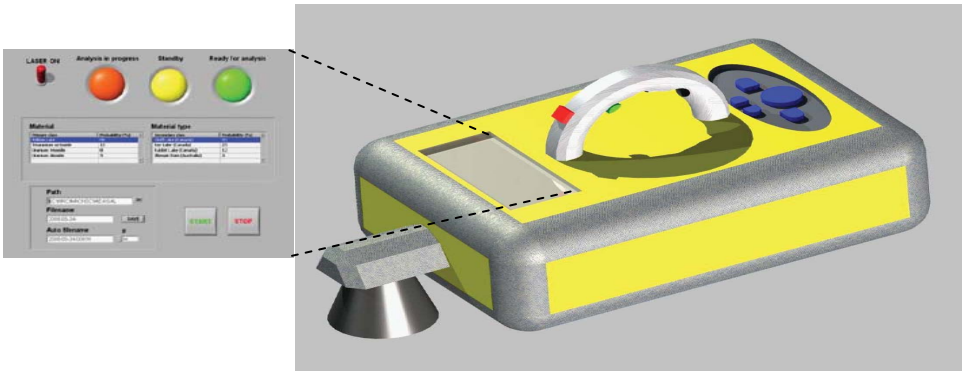

Figure 11: Conceptual HLD

Conclusion

The handheld LIBS device shows great promise as a useful and versatile tool to enforce nuclear safety and security. This instrument will enable IAEA inspectors to instantaneously identify

materials of interest during any kind of inspection, and will also minimize the number of “normal” swipe samples analyzed by SAL and IAEA laboratories. This capability will allow results to be returned to inspectors much more rapidly than in the past, due to their discrimination of swipe samples in the field. Beyond the IAEA, border services officers can use the HLD to identify or confirm suspicious material. Moreover, it is expected that first responders and other service providers will find additional uses for this HLD and LIBS technology, where the LIBS instrumentation is limited only by the number of material spectra in the system’s database.

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Standoff Performance of HPGe Detectors in Identification of Gamma-Ray Radiation Sources

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Abstract:

The detection and identification of radiation sources at distances in the range of 15 meters or more is becoming increasingly important for illicit materials interdiction and the location of lost or orphan sources. In most locations, there is a considerable gamma-ray flux from natural background (NORM) and cosmic-induced nuclides. This gamma-ray flux varies with time, weather conditions, location, and changes in the materials at a location such as a portal. All of these contribute to changes in the detector total count rate unrelated to the nuclides of interest and can mask considerable quantities of material. The high resolution of HPGe enables the nuclide-specific peak and background counts to be extracted from the spectrum of the suspect object or area without relying on background spectrum subtraction techniques using background spectra necessarily collected at either different places or times. Data were collected with and without collimators. The collimators reduce the detector field of view and improve the signal-to-noise ratio thereby reducing the minimum identifiable activity (MIA). A straightforward summing technique allows the data from multiple detectors to be aggregated to improve the signal. Data have been collected for ^{137}Cs at distances up to 80 meters and used to predict the performance at 100 m. The MIA has been calculated for given false positive and false negative rates for systems with up to 8 HPGe detectors.

Keywords: Standoff, HPGe, illicit trafficking, detection limit

1. Introduction

Long distance or standoff measurements are increasingly important for illicit materials interdiction and the location of lost or orphan sources as the operations move from portal monitoring to searching. The search for radiation sources can take place where the distance between the detectors and the sources is in the range of 15 meters or more. In addition, it is necessary to identify the nuclides producing the gamma-ray flux as well as detect the increase in gamma rays. In most locations, there is a considerable gamma-ray flux from natural background (NORM) and cosmic-induced nuclides which can mask the material of interest. This gamma-ray flux varies with time, weather conditions, location, and changes in the materials at a location such as a portal as well as normal variation during a search over a wide area [1]. All of these contribute to changes in the detector total count rate unrelated to the nuclides of interest and can hide considerable quantities of material. The good resolution of High Purity Germanium Detectors (HPGe) enables the nuclide-specific peak and background counts to be extracted from the spectrum of the suspect object or area with sufficiently low uncertainty to make nuclide identifications on fewer total peak counts than low resolution detectors. [2] Because of the constantly changing background, this is best done without relying on spectrum subtraction techniques using background spectra necessarily collected at either different places or times. The HPGe data were collected with and without collimators on the detectors. The collimators reduce the detector field of view and improve the signal-to-noise ratio thereby reducing the Minimum Identifiable Activity (MIA). A straightforward summing technique allows the peak and background data from multiple detectors to be aggregated to improve the signal. The combination is not done on a channel-by-channel basis. Data have been collected for ^{137}Cs at distances up to 80 meters and used to predict the performance at 100 m. The MIA has been calculated for given false positive and false negative rates for systems with up to 8 HPGe detectors.

2. Equipment and Setup

HPGe Detectors
The HPGe detectors used were ORTEC IDMs, as shown in Fig. 1. Each IDM is a fully integrated gamma spectrometry subsystem consisting of an 85 mm diameter x 30 mm deep, p-type HPGe crystal, Stirling cooler, DSP MCA, high voltage supply, shielding against gamma rays from behind the front surface, and high speed USB communication. A complete description is given in [3]. The large diameter detector is optimized for energies in the 100 to 400 keV range, which is important for detection of SNM.

Eight IDMs were used for this measurement. The relative efficiency according to the IEEE 325 method ranged from 50 to 55%.

Figure 1 The Interchangeable Detector Module

Mounting
The eight IDMs were mounted in 2 m high cabinets with 4 IDMs in each cabinet. The IDMs were uniformly spaced in the vertical direction. The cabinets were positioned side-by-side as shown in Fig. 2.

At the distances measured, the precise relative positions of the IDMs do not impact the resulting data. The data were collected in a typical factory-type building with a concrete floor, gypsum internal walls, and steel supported roof. The data were collected in list mode. List mode enables the data to be combined in many different ways after collection, such as different integration times.

The background flux is incident from all directions. The ^{137}Cs flux was incident from the front only.

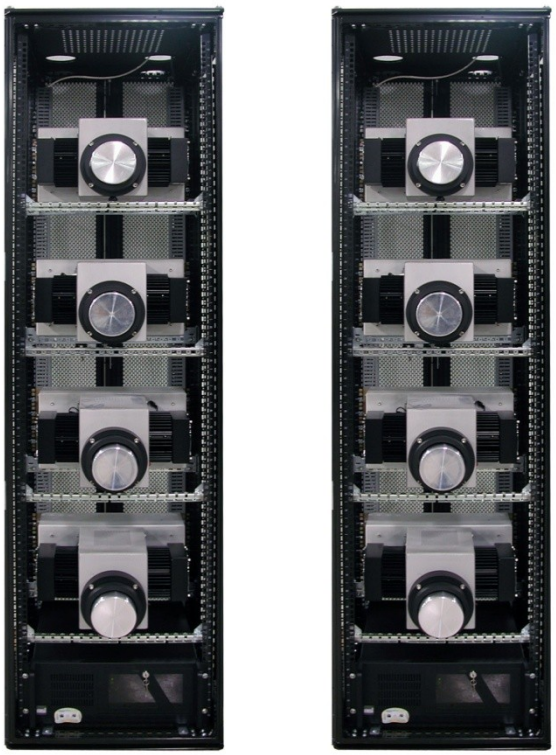

Figure 2 Eight IDMs in two cabinets

Shielding
The IDM includes some steel shielding (the black ring around the detector endcap in Fig. 2). Data were collected in this configuration. It is also possible to add additional shielding on the sides of the detectors to reduce the background contribution from the sides and to reduce the field of view. Cylindrical shielding can be placed on the endcap in front of the black shield to reduce the contributions from below (nearby ground) and above (buildings or sky shine).

The steel shielding extends from just behind the detector crystal for a distance of 10 cm. It is 12 mm thick for 4 cm and 25 mm thick for the remaining length. The additional shielding was 5 cm of lead for the vertical side shield and 12 mm of lead for the cylindrical shield. The side shield extends from the steel shield outer diameter to 13 cm in front of the detector endcap. The cylinder shield extends from the steel shield inner diameter to 4.5 cm in front of the detector endcap. This is shown in Fig. 3.

Peak Quality Factor
The identification of the nuclide is based on the presence in the spectrum of the intense gamma rays from that nuclide. The peak is present in the spectrum if a measured parameter is above a threshold. The peak parameter or peak quality (Q) is defined as the net peak area divided by the uncertainty in the net peak value [4]. The threshold is based on the desired false positive rate and false negative rate. There is a threshold specified for each gamma ray in the analysis table. The gamma rays in the table are given in [5, 6, and 7].

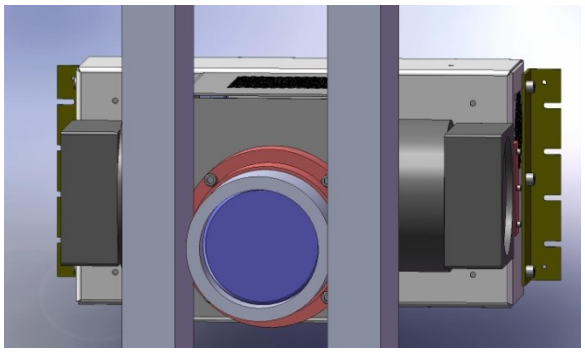

Figure 3 IDM with side and cylindrical shielding

The Q can be calculated for the 8-IDM combination data in a similar manner, thus giving the Q for a detector of 8 times the efficiency of a single IDM.

Source
The source was a ^{137}Cs point source of 0.46 mCi at the time of measurement. It was positioned at 1 m from the floor on a ring stand with little material near the source. The source was positioned in front of the IDMs at 10 m intervals. The length of the room limited the maximum distance to 80 m.

3. Field of View

The Field of View (FOV) is the area in front of the detectors where a source of gamma rays (NORM or other source) could contribute to the spectrum. It can be expressed in angle or length at a distance (between the emitter and detector). The effective FOV is the area in front of a detector where a source could contribute significantly to the spectrum. The effective FOV can be much smaller than the actual FOV because the source contribution is limited by the reduction in flux due to distance ($1/r^2$) and absorption by the air. Background activity outside the effective FOV should be blocked from the spectrum by collimation.

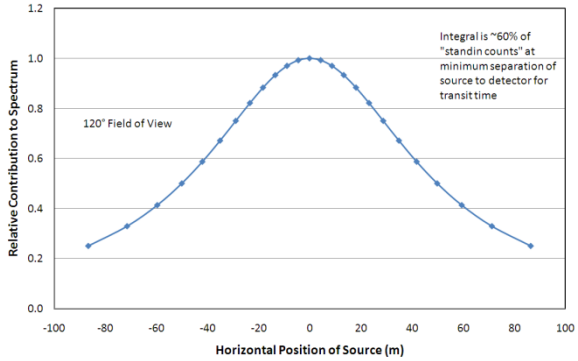

Figure 4 Contribution to the Spectrum for a Source at 50 m

Figure 4 shows the relative contribution to the spectrum of a source at positions along a line that is 50 m from the source at its closest position. The contribution to the spectrum is normalized to the contribution when the source is at the minimum distance. In a measurement where the source is moving relative to the detector, either searching or portal monitor, there is little relative contribution to the spectrum for large horizontal distances. If the source is moving, the contribution to the spectrum in a 120º FOV is about 60% of the contribution of a source stationary at the minimum distance, ignoring air attenuation.

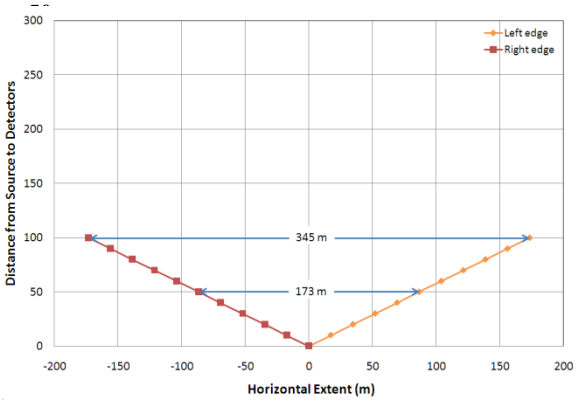

Figure 5 Width of 120º FOV at various distances

Figure 5 shows that the source could contribute to the spectrum over a length of 173 m when the FOV is 120º (as defined by the collimation) the minimum source-to-detector distance is 50 m.

Comparing Figs. 4 and 5, it can be seen that the background contribution could be reduced without a reduction in the source contribution with shielding. Thus, the signal-to-noise ratio could be improved by reducing the FOV with collimation.

4. Background

Figure 6 shows the background from one IDM with only the integral steel shield in place. The total count rate is about 100 cps. The largest peak is ^{40}K with a count rate of 1.06 cps. There are few counts above 2 MeV. The distribution with energy is typical of background spectra. Most of the counts are in the region below 250 keV and are scattered gamma rays. This energy region is important for SNM detection.

The spectrum for the same detector with the addition of the 5 cm lead side shields is shown in Fig. 7. The total count rate (0 – 3 MeV) is about 50 cps and the ^{40}K count rate is 0.45 cps. This reduction in background by a factor of about 2 will result in a reduction of the MIA by about 1.4. Note that the region below 250 keV is reduced, indicating that this region does not contain many counts from Compton scattering inside the detector.

The above spectra are typical, but the background varies significantly with time. Figure 8 shows the background for the peak region at 661 keV with no ^{137}Cs present and the side shields installed. The average background is 41.6 counts with the minimum at 19 and the maximum at 67. At 10 kph, the source will be in the FOV for about 62 s at the minimum separation distance of 50 m. In the following results, the peak analysis is done using the background under the peak in the actual spectrum rather than using a stored background.

5. Results

The spectrum of ^{137}Cs, positioned at 20 m for the sum of 8 IDMs and a data collection of 20 s is shown in Fig. 9. This is without the side shields. The peak analysis does not use this summed spectrum, but rather sums the peak results from each IDM. This method reduces the need for precise channel alignment and preserves the resolution of each detector in the result.

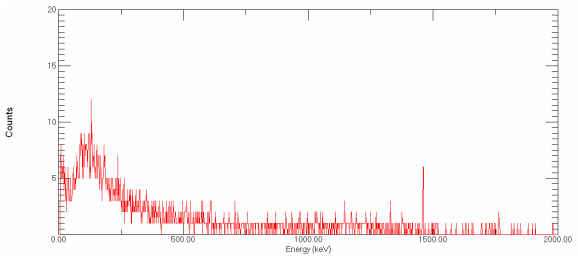

Figure 6 Background with no Extra Shielding (60 s)

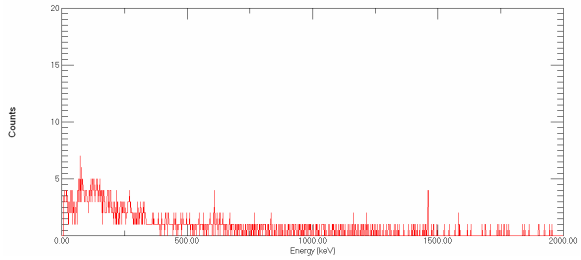

Figure 7 Background with Vertical Side Shielding (60 s)

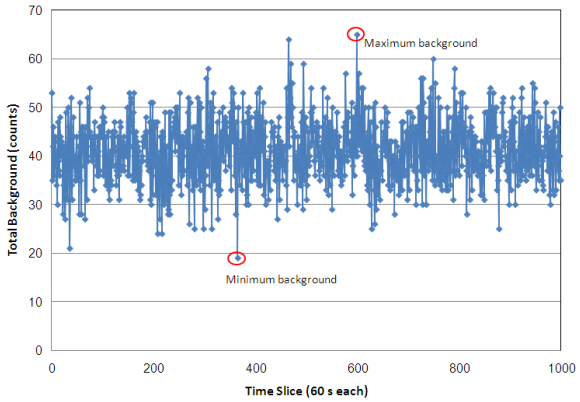

Figure 8 Background Variation with Time in the 661 keV peak region in 8 IDMs

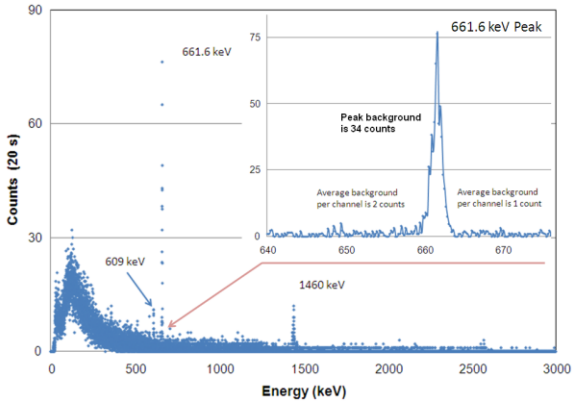

Figure 9 Spectrum of ^{137}Cs at 20 m for 8 IDMs Collected for 20 s without side shields

Figure 10 shows the same spectrum for the 50 m source position. Note that the net peak area is impacted by both the distance and the absorption by air.

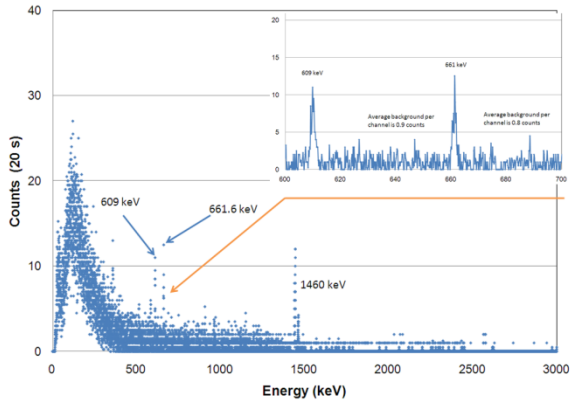

Figure 10 Spectrum of ^{137}Cs at 50 m for 8 IDMs Collected for 20 s without side shields

Figure 11 shows the measured count rate of ^{137}Cs at distances of 10 to 80 m. For comparison, the background peak areas at 609, 1461, and 2614 keV are shown. The $1/r^2$ is also shown. Note that for distances above 30 m, the ^{137}Cs peak rate is below the background peak count rate without side shields. Figure 10 shows the 609 and 661 keV peaks clearly separated. For low and medium resolution detectors, these peaks will merge into one peak. In addition, the variation in the background (see Fig. 8) by a factor of 3 over time means that good resolution is the only way to detect low activities.

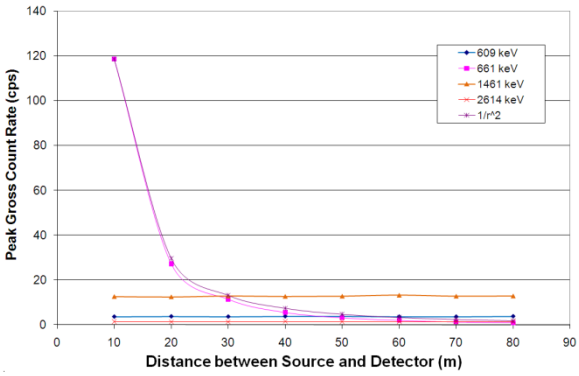

Figure 11 Count rate for 137Cs using 8 IDMs vs Distance

The MIA is related to the Q for the peak or peaks of the nuclide in question. The relationship is the peak area: Q must be above the threshold for the peak to be present, and the MIA is the activity that would produce that peak area. Figure 12 shows the Q for different distances with and without the side shielding. The Q is improved by about 18% with the shields at a source distance of 80 m. Previous work showed that for most cases, a threshold of 5 for Q will meet the 1:10000 FP and the 1:1000 FN rates. At 100 m distance, the extrapolated Q value is 5.1, indicating that an 8 IDM system with shielding would be able to detect 1 mCi of ^{137}Cs in 60 s.

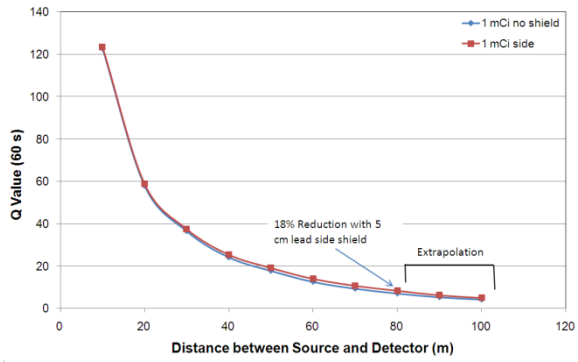

Figure 12 Q Based on 8 IDMs for Different Shielding

This same data can be used to estimate the time for an eight IDM system with shielding to detect 1 mCi at various distances as shown in Fig. 13, when the threshold is 5.

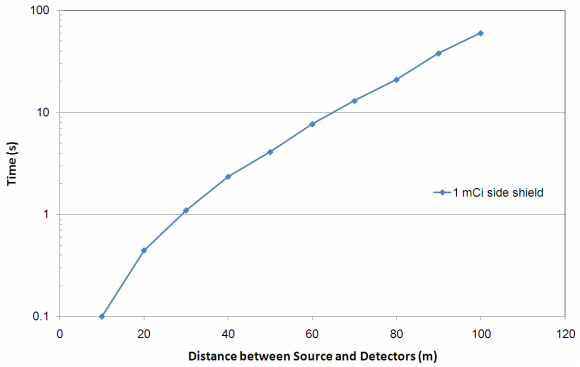

Figure 13 Time to Detect by Distance for 8 IDMs

6. Conclusion

These measurements with an 8 IDM system with the internal steel shield and 5 cm of lead shield show that a 1mCi ^{137}Cs source can be detected with conservative FP and FN rates at a distance of 100 m in about 60 s. In special circumstances, for example, when additional information suggests the presence of SNM in a certain location, the search operation may be willing to accept a higher FP rate (by lowering the Q threshold) to improve sensitivity. Lowering the Q threshold will substantially reduce the ^{137}Cs identification time. This result depends on the background in the spectrum at 661 keV from both the natural background and any other sources that may be present. With suitable collimation to reduce the background, this time is still within the expected time a source would be in the FOV for a search system moving at 10 kph.

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Neutron measurement techniques for hand-held devices

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Abstract:

Reliable measurement techniques are important for early detection of illicit use of nuclear and radioactive material. Nuclear safeguards are an important tool to impede the illicit handling of fissionable material. But the scope of safeguards is only nuclear material within the fuel cycle. Neutron emitting radioactive material may have its origin from the fuel cycle as well as from an industrial neutron source. Terrorist attacks using neutron emitting material may have far reaching consequences not only if nuclear material is involved. Therefore it is of great importance that mobile measuring systems are available to detect and to respond to possible malicious acts involving this material.

Fraunhofer INT has built up a mobile measuring system in a transportable container for the detection and identification of radioactive and nuclear material. The system is equipped with various types of detectors for neutron and gamma radiation, some of the detectors are similar to those used in safeguards. The measuring capabilities of the system have been extended by mobile, easy transportable neutron measuring devices. These devices not only count neutrons but are also able to give additional spectral, coincidence, or multiplicity information. Thereby different types of neutron sources, like SNM or industrial, can be distinguished. Various mobile neutron detection devices have been tested in different neutron fields of variable strength and with additional shielding of different types.

Results of these measurements will be presented. The advantages and disadvantages of different concepts will be shown. This may help to prevent illicit trafficking of nuclear material.

Keywords: illicit trafficking, neutron detector, in-situ measurement, nuclear terrorism, dirty bomb

1. Introduction

The threat of terrorist attacks with dirty bombs which include radioactive or even nuclear material is a widely discussed topic. Dirty bombs may have far reaching consequences not only induced by the radioactive or nuclear material involved but also by the conceivable disturbance of public and private life. In the case of such a threat it is very important to know which type of radioactive or nuclear material is involved and how much of it is present to determine the possible effect of the dispersion of radioactive material. Therefore sophisticated measurement devices suitable for in-field use are crucial.

Whereas gamma measuring devices are widely used neutron devices are less common. In some cases neutron measurement is very important. For example, in the case of a highly shielded device where gamma radiation can not pass through the shielding material, neutrons may still be detectable. In addition, neutrons have the advantage that the neutron background is generally low.

Neutron measurement is especially important in case nuclear material is involved. Then the crucial question for the task force will be: could it not be an improvised nuclear device rather than a “simple” dirty bomb? Novel portable neutron measuring devices are now available which address this question in different ways.

Fraunhofer-INT has built up a transportable measurement system for the detection and identification of nuclear and radioactive material, integrated into a transportable

container [1]. The container is shown in figure 1. This system is completed by a car equipped with measurement systems to detect and locate such material.

Figure 1: Transportable container, gamma and neutron measuring devices in front. On the right side power generator on separate trailer.

The novel hand-held neutron measurement devices investigated in this work are also suitable to enhance the detection capabilities of this measurement system.

2. N-probe

If neutron sources are detected and the information of the existence of neutron sources alone is not sufficient an identification of the type of source is necessary. This can either be done by obtaining an energy spectrum or by doing a coincidence measurement. The latter will distinguish between industrial neutron sources and special nuclear material. Generally neutron spectrometers are systems which have to be used in laboratories. The spectroscopic neutron measuring device N-probe, manufactured by BTI, is an exception (see figure 2). This is a portable small and rather light-weight battery operated instrument which is able to determine a neutron spectrum in the field with sufficient energy resolution. The dose rate can be in the range of up to 200 μSv/h.

The probe has a weight of 4.1 kg and can be carried by the handle easily while the control unit is carried with a shoulder strap. With the associated small hand-held computer the results for the neutron flux, fluence, dose rate, accumulated dose and the neutron energy as well as the dose spectra are calculated and displayed on the screen. The battery lifetime is about 12 hours.

Figure 2: N-probe neutron spectrometer. The hand-held computer placed in the front is displaying the measured energy spectrum. A neutron source is placed on the little lifting plate. The N-probe is behind the computer with the two detectors on the left side, liquid scintillator at the top and ^{3}He tube below.

With the N-probe spectroscopy from thermal up to a neutron energy of 18 MeV is possible. For this purpose two separate detectors are used. The energy region from thermal up to 800 keV is covered with a ^{3}He tube, a liquid scintillator is used to cover the region from 800 keV to 18 MeV.

Figure 3 shows the comparison of energy spectra obtained from three different neutron sources, an Am/Be source with a neutron emission of 207000 n/s, which was placed in a distance of 25 cm, an Am/Li source with a neutron emission strength of 54000 n/s, which was placed in a distance of 5 cm, and a ^{252}Cf source with a neutron emission of 16000 n/s, which was placed in a distance of 5 cm. It is clearly possible to distinguish between the three sources.

The neutrons generated by the Am/Be source typically have a mean energy of ~ 4-5 MeV. The Am/Li source provides neutrons with a maximum energy somewhat above 1 MeV which is a significantly lower energy compared to Am/Be and ^{252}Cf sources. The mean energy of ^{252}Cf is between the mean energies of neutrons from Am/Be and Am/Li sources. It is ~ 2-3 MeV. These facts can clearly be confirmed in the obtained spectra.

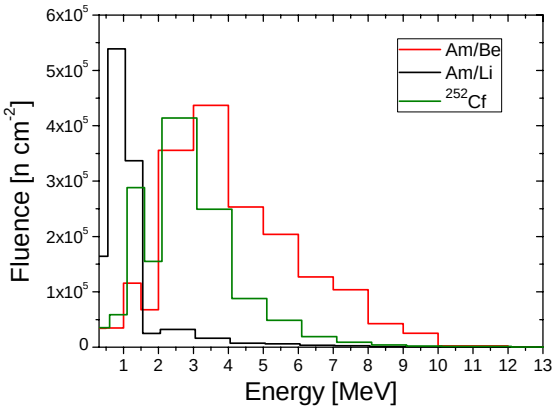

Figure 3: Energy spectra for different neutron sources. Am/Be (207000 n/s, distance 25 cm), Am/Li (54000 n/s, distance 5 cm) and ^{252}Cf (16000 n/s, distance 5 cm). Measurement time was 72000 s. The lines are slightly shifted against each other along the energy axis for better illustration.

Figure 4: Fission Meter panel unit.

3. Fission Meter

The Fission Meter, manufactured by Ametek/ORTEC, is a portable, battery-operated system for the identification of neutron sources. It features 30 ^{3}He tubes in two panels in a hinged array. In this way a suspicious object can be surrounded to further increase the system's efficiency. On one side of the panels moderator material (polyethylene) was fitted to measure fast neutrons. The system's computed output signal enables the user to discriminate fission material from industrial neutron sources by the presence of coincident neutrons. Fission material emits between 2 and 5 coincident neutrons per fission process, industrial sources only one neutron per reaction. In this way a characterization of present nuclear material is possible with the Fission Meter [2].

Figure 4 shows the Fission Meter system. The unit may be placed in a bag and can be carried around this way. Because of the relatively high weight of 26 kg an extended search is strenuous. The panel unit features a display showing the neutron count rate. It is connected by a serial cable to the analysis unit called Ranger (yellow/black device on the right of figure 5). The Ranger contains a pocket PC which can be remotely controlled on a laptop computer via USB connection with the "Active Sync" software which also serves for transferring files from the Ranger to the laptop. The laptop screen shows an enlarged display of the Ranger screen. Table 1 shows an overview of characteristic data of the Fission Meter.

Figure 5: Pocket computer control unit ("Ranger") and remote display on laptop screen, data transfer via serial cable.

Gas characteristics	^{3}He (7.6 · 10^{5} Pa)
Diameter per tube	2.54 cm
Length per tube	48.26 cm
Number of tubes	15 per panel => 30 per device
Active area (panel)	~1800 cm^{2} (15 tubes)
Moderator	Polyethylene, on one side (minimum 2.54 cm)
Weight	26 kg (57 lbs)

Table 1: Characteristic data of the Fission Meter.

The system offers three different modes of operation:

- Mobile Search Mode
- Static Search Mode
- Characterization Data Collect Mode

The mobile search mode displays temporal changes of the neutron count rate and is therefore suitable for locating nuclear or radioactive material during on-site inspections. The static search mode and the characterization data collect mode may be selected to measure the exact count rate once a suspicious object emitting an elevated neutron radiation field is located. The latter mode allows determining the presence of coincident neutrons by means of multiplicity plots which leads to the conclusion that special nuclear material (SNM) such as ^{235}U or ^{239}Pu is involved. If these plots show a Poisson distribution, neutrons were emitted randomly. If there are significant differences between a multiplicity distribution and the corresponding Poisson distribution, coincident neutrons and therefore SNM was present. The analysis routine also features plots of the Feynman variance which are meant to give clues if the nuclear or radioactive material was surrounded by shielding material.

Figure 6 shows the exemplary result of a mobile search of an Am/Li source (54000 n/s) with different distances between source and detector. The source could clearly be located up to a distance of 1 m.

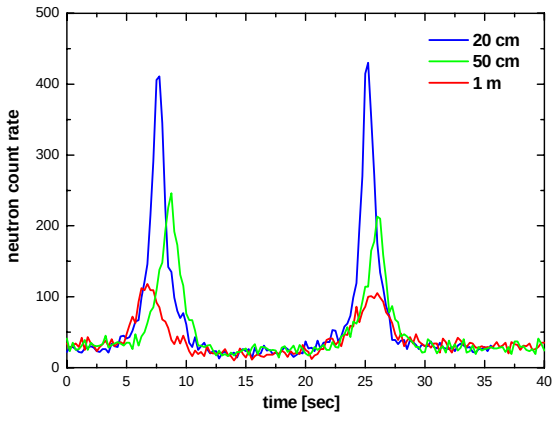

Figure 6: Localization of an Am/Li source (54000 n/s) with the mobile search mode (in the lab).

Figure 7 and figure 8 illustrate the differences of neutron multiplicity plots of a ^{252}Cf source (22000 n/s), which emits coincident neutrons (representing SNM), and the Am/Li source mentioned above, which does not emit a relevant yield of coincident neutrons. In the case of the former, the multiplicity distribution clearly differs from a Poisson distribution (see figure 7). In the case of the latter, the multiplicity distribution is almost identical to a Poisson distribution (see figure 8). This way SNM can be distinguished from radioactive material emitting neutrons at random.

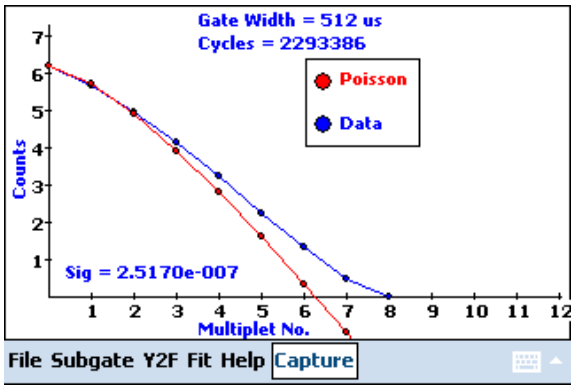

Figure 7: Multiplicity plot of a ^{252}Cf source (22000 n/s), the measuring time was approximately 20 minutes.

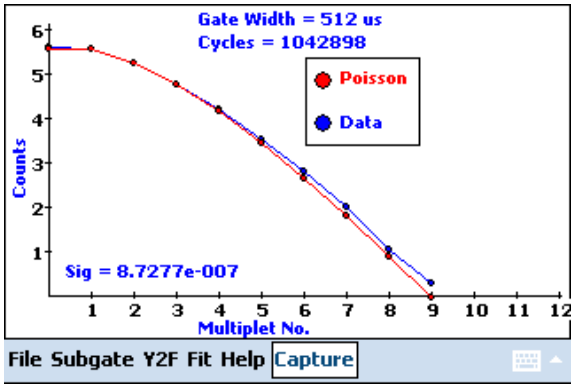

Figure 8: Multiplicity plot of an Am/Li source (54000 n/s), the measuring time was approximately 20 minutes.

4. PUMA

Most transportable measuring systems for neutron detection use ^{3}He as detector material. Despite the many advantages of ^{3}He there are some significant limitations. The PUMA detectors from NUCSAFE represent a suitable alternative. The PUMA neutron detectors use a glass fibre scintillation detector with incorporated ^{6}Li as neutron-active material for the detection of neutrons. This type of detector provides some important advantages compared to ^{3}He detectors, one of them being the ability

to build sensors with large areas up to 5 m^2. Another advantage is the high density of the glass fibre detections system compared to the gaseous ^{3}He which results in a better detection efficiency for the same detection volume. The use of high-speed electronics which is hardly possible with gas detectors is a further advantage of the PUMA detectors. Because the glass fibres are sensitive to both neutrons and gammas the detection signals have to be separated via neutron to gamma discrimination.

The Guardian PRST (Portable Radiation Search Tool) is a portable PUMA detector designed as a suitcase (see figure 9). The suitcase has the dimensions of 50 cm x 36 cm x 10 cm with a detection area of 500 cm^2. Because of its unsuspicious appearance it is convenient for covered search and surveillance tasks. Battery operated and containing a PC104 microprocessor all of the operations can be controlled without an external PC. Hence the Guardian PRST combines high sensitivity for neutron detection with portability and allows the user a high flexibility in operation.

Figure 9: Picture of the Guardian PRST.

The pulse discrimination separates the measured counts into a gamma and a neutron channel. By changing the parameters of the discriminator it is possible to vary its effectiveness, but with the consequence that the neutron efficiency (counts in the neutron channel) is changed too. In an area with increased gamma radiation, with its necessity to obtain the best neutron to gamma discrimination, discrimination rates over 1:8500 are possible. For example, in this case the neutron efficiency falls bellow 2 % for ^{252}Cf. If high neutron efficiency is required (above 10 % is possible) then the discrimination effectiveness drops to 1:5 for ^{252}Cf.

The detector provides the measured count rate as online measuring result. In figure 10 a measurement obtained with a ^{252}Cf source is displayed for different source distances as an example. The source had a neutron emission rate of 16000 n/s. In figure 11 the measuring configuration is shown for a distance of 50 cm.

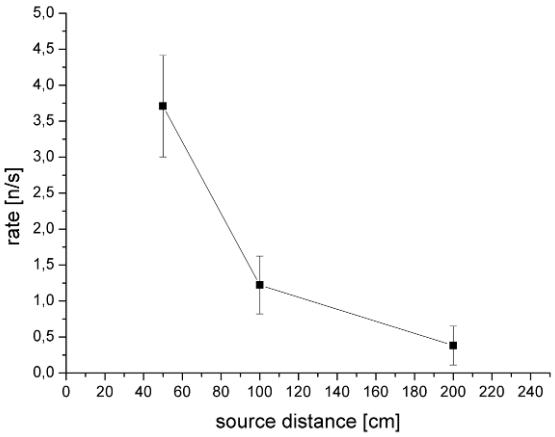

Figure 10: Count rate measurement results for a ^{252}Cf source (16000 n/s) for different source distances.

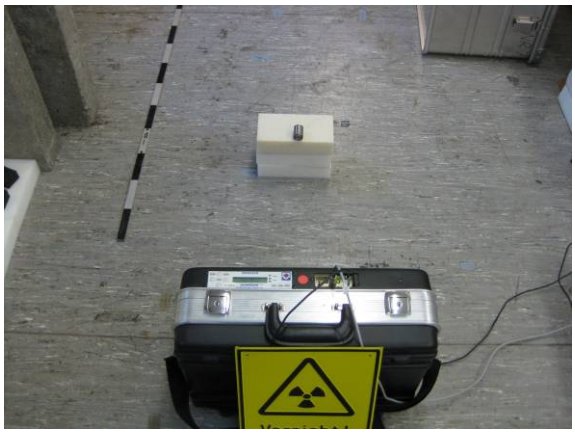

Figure 11: Measuring configuration for a distance of 50 cm.

5. Conclusion

Three hand-held neutron measurement devices with enhanced capabilities were investigated. The N-probe is a light-weight portable instrument facilitating neutron spectroscopy which can be used in the field. Although the spectroscopic information is of limited energy resolution it can be used to distinguish different types of neutron sources. The Fission Meter is a transportable neutron multiplicity counter with high efficiency. Its software allows

differentiation of fission sources from random neutron sources. The PUMA device is a portable neutron counter based on a lithium-glass detector with high neutron efficiency and reasonable gamma discrimination.

Neutron measuring equipment which formerly needed laboratory set-up is now available as compact, light, portable systems usable in the field. Thus it is now possible to detect and classify neutrons in addition to gammas in-situ. Therefore the nuclear signature of an unknown object containing radioactivity which might be a dirty bomb or even an improvised nuclear device can be detected and evaluated comprehensively and the possible harm can be estimated more precisely. Therefore, these systems assist in preventing nuclear terrorism.

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Feasibility Study of a Portable Coupled ^{3}He Neutron Detector with $LaBr_3$ Gamma Scintillator for the Purpose of Plutonium Identification and Quantification

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Abstract:

In recent years there has been several research endeavors to increase the ability to identify and quantify special nuclear material in field measurements. Many have included both gamma spectroscopy and neutron coincidence systems that are portable and work in a variety of environments. In this work a MCNPX[1] model was designed that includes four gamma detection slabs placed within four neutron detection slabs. Four Plutonium (Pu) samples of known quantity were modeled and tested to determine what data was available from each individual signature. Each model included a separate MCNPX deck for each individual isotope that contributes to the gamma signature in photon mode and a spontaneous fission and (α,n) deck for the neutron signature. The first three samples were used to create spectrums and efficiency curves for each odd isotope as well as for a Pu effective mass for the neutron signature. The data from these simulations were then used to identify the isotopics in the fourth sample to within acceptable accuracy. From this data a total Pu mass was obtained as well as an ability to determine the ratio of (α,n) to spontaneous fission neutrons without additional simulations. This provides a new method to detected and identify the Pu content within a sample without producing additional information since isotopics can be determined with the use of the gamma and neutron system.

Keywords: neutron coincidence counting, gamma spectroscopy, plutonium identification and quantification

1. Introduction

Neutron coincidence counting has been available for many years and is a well established method to quantify special nuclear material. However, there are limitations to its applications. The most notable of these is the requirement to know the isotopics of the sample being measured. The purpose of this work was to study the advantages of using a coupled neutron and gamma measurements in a single field deployable detector system over currently available portable neutron coincidence counters. The system of interest should be portable so that it can work in a number of environments. It should also have a small foot print to minimize the space required. Coupled neutron and gamma measurements have been studied previously[2,3]; however, this work focused on a portable system using an advanced γ-spectroscopy system[4] which would be field deployable. The feasibility of this design concept was studied using MCNPX simulations.

2. Theory

The material of interest in this work is special nuclear material but we will focus primarily on PuO_2. PuO_2 emits neutrons due to spontaneous fission (SF) and (α,n) reactions. It also emits characteristic γ-rays that are produced in coincidence with α-emissions. The neutrons and gammas travel at different speeds and therefore are detected at different times within the detector.

2.1 Coincidence counting

There are several different types of coincidence counters available today of various shapes, sizes, and efficiencies. Many consist of a slab or well design and operate with efficiency generally above 20%, but most of these systems are not easily movable. They tend to be heavy and cumbersome and are not able to be used in the field.

A coincidence counter determines Pu mass in the sample by measuring the singles or totals and doubles or reals (coincident) neutrons that are produced by (α,n) and spontaneous fission. The totals (T) and reals (R) count rates are given by the so-called Neutron Coincidence Point Model[5]:

$$T = m^{eff}_{240} Y_{240} \varepsilon M \nu_{s1} (1 + \alpha) \quad [1]$$

$$R = m^{eff}_{240} Y_{240} \varepsilon^2 \frac{\overline{\nu(\nu - 1)}}{2} F \quad [2]$$

$$\overline{\nu(\nu \quad 1)} = M^2 \left[\frac{\nu_s(\nu_s - 1)}{1 + \alpha\nu_s} + \frac{M - 1}{\nu_i - 1} \frac{1 + \alpha}{1 + \alpha\nu_s} \nu_s \nu_i \overline{(\nu_i \quad 1)} \right] \quad [3]$$

where m^{eff}_{240} is the $^{240}Pu_{eff}$ mass, Y_{240} is the specific spontaneous fission rate for ^{240}Pu (in fissions/sec/g), F is the doubles gate fraction, ε is the detector efficiency, M is the sample self-multiplication, α is the ratio of neutrons produced from (α,n) reactions to those from spontaneous fission reactions, and $\nu_s(\nu_s\text{-}1)$ and $\nu_i(\nu_i\text{-}1)$ are the factorial moments of the spontaneous fission and induced fission neutron distributions.

The ^{240}Pueff mass is the exact mass of ^{240}Pu that would create the same totals and reals count rate as would be measured from the actual sample. The ^{240}Pueff mass (m^{eff}_{240}) is given by:

$$m^{eff}_{240} = 2.52m_{238} + m_{240} + 1.68m_{242} \quad [4]$$

where m_{238} is the ^{238}Pu mass in the sample, m_{240} is the ^{240}Pu mass, and m_{242} is the ^{242}Pu mass. The total Pu mass is given by

$$m_{Pu} = m_{238} + m_{239} + m_{240} + m_{241} + m_{242} + m_{Am241} \quad [5]$$

where m_{239} is the ^{239}Pu mass in the sample, m_{241} is the ^{241}Pu mass, and m_{Am241} is the ^{241}Am mass. Thus to get the total Pu mass the isotopic ratio of the Pu is needed. For an unknown sample in the field, this quantity would not necessarily be known. The ε, F, $\nu_s(\nu_s-1)$ and $\nu_i(\nu_i-1)$ are known for samples of unknown mass and isotopics. The values for m^{eff}_{240}, M, and α are unknown. This leaves two equations with three unknowns. Therefore, at least one of the unknowns must be determined by an alternate means. Typically M or α are calculated on assumptions about the sample isotopics and/or geometry.

2.2 Gamma detection

There are many different types of γ-spectroscopy systems currently used. The two main ones are solid-state and scintillation systems. They both have advantages and disadvantages. High Purity Germanium (HPGe) is the most common solid-state detector available and has the best resolution available on the market today. This resolution comes at a cost of portability since it has to be kept at liquid nitrogen (LN) temperatures for proper function. There are portable systems available but these have some limitations.

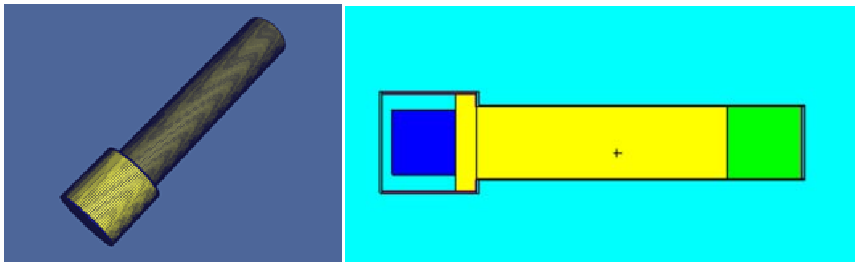
Scintillation detectors, most commonly Sodium Iodide (NaI)[7], are generally much more portable than HPGe but have a lower resolution. They are also much more rugged, can operate at room temperature, have higher efficiency, and are available in larger crystals than HPGe. In recent years, there has been a tremendous amount of research to improve the resolution of scintillation detectors. One of the most notable advancements is the use of Lanthanum Bromide ($LaBr_3$) crystals. $LaBr_3$ has superior resolution to NaI (though still less than HPGe). Also in recent years, the development of nanocrystals embedded in a clear matrix has been developed at LANL using $LaBr_3$ crystals as the scintillation material[4]. This should be able to increase the available crystal size and provided alternate variations in geometry.

3. MCNPX benchmarking

Since nanocrystals detectors are still under development and unable to be obtained, an alternate system was modeled in order to validate the usage of MCNPX for $LaBr_3$ crystals. For this, the Canberra IPROL-1 probe (Figure 1) that works with the Inspector 1000 portable gamma spectroscopy system was chosen. The probe was modeled in MCNPX. The model included the crystal, crystal housing, photomultiplier tube, electronics portion, and casing. This model can be seen in Figure 2. The MCNPX simulation included an F8:p pulse height tally with a 4000 bin energy grid to simulate a pulse height spectrum. A Gaussian

Energy Broadening (GEB) function[1] was included to appropriately broaden the peaks with FWHM values versus energy measured using a number of sources.

Figure 1. Image of IPROL-1 $LaBr_3$ probe



(A) (B)

Figure 2. MCNPX model probe (A) 3-D visualization (B) cross-sectional view

The spectrum from a ^{137}Cs standard was acquired using the IPROL-1 probe. This experiment was then simulated using the MCNPX model. Figure 3 shows the measured and the MCNPX simulated spectra for the ^{137}Cs source. As can be seen, the ^{137}Cs photopeak agreement between the measured and simulated results is generally quite good. However, the agreement in the Compton background and the peaks at 1.435 MeV is poor. This is due to the "internal" background in the detector due to its natural radioactivity.

Natural La contains a small amount of ^{138}La which decays by electron capture with a 1.435 MeV gamma 66.40% of the time and by beta (β) decay with a 788 keV γ and a 252 keV β 33.60% of the time. When both modes of decay are included the 1.435 MeV peak was simulated correctly. The broad plateau between 788 keV and 1040 keV was due to the β particle and the 788 keV γ being detected in coincidence. The β decay produces a continuous energy spectrum; this in essence will nonsymmetrically broaden the 788 keV photopeak. The gamma can also undergo Compton scattering causing the lower energy γ to be detected in coincidence with the β. If the γ escapes entirely just the β detected. These possibilities were accounted for in the simulated spectrum by including the continuous β spectrum in the probability of detection produced by the output of the F8:p tally for every energy bin from E_{bin} to $E_{bin+252\ kev}$ until 788 keV plus 252 kev[7,8,9]. This produced a spectrum that broadened the 788 keV peak and continuum to better represent the actual spectrum. The corrected spectrum is labeled "Corrected MCNPX" in Figure 3.

To confirm the proper simulation of the impact of the La internal radioactivity on the spectrum, a measurement and a simulation of the background only (i.e. without a source present) was performed. The measured and "Corrected MCNPX" spectra for this background is shown in Figure 4. These results were used to develop the procedure for the modeling the $LaBr_3$ detector in the coupled neutron-gamma detector concept being considered here.

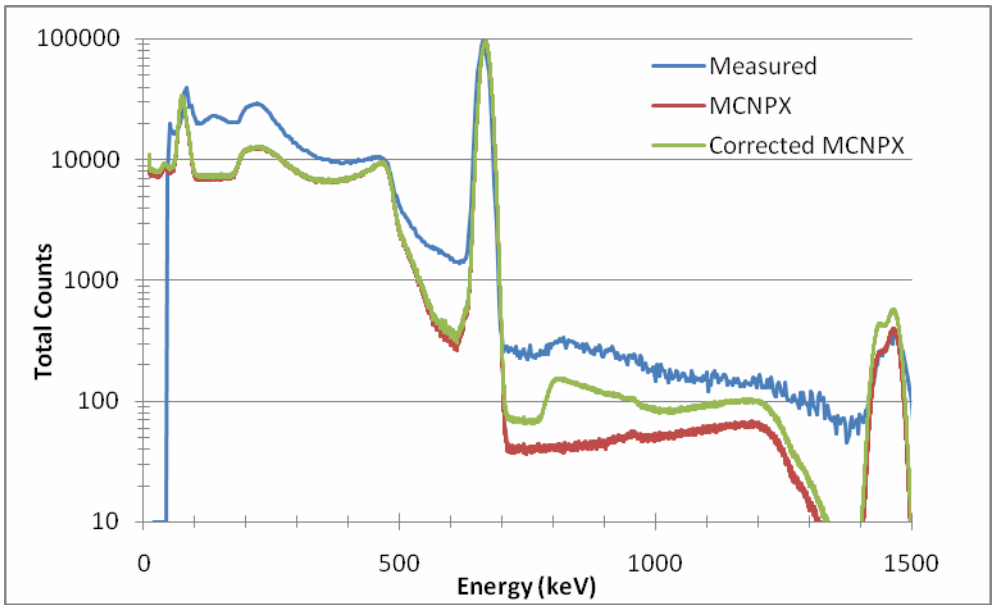


Figure 3. Measured and simulated spectrum for ^{137}Cs for $LaBr_3$ probe

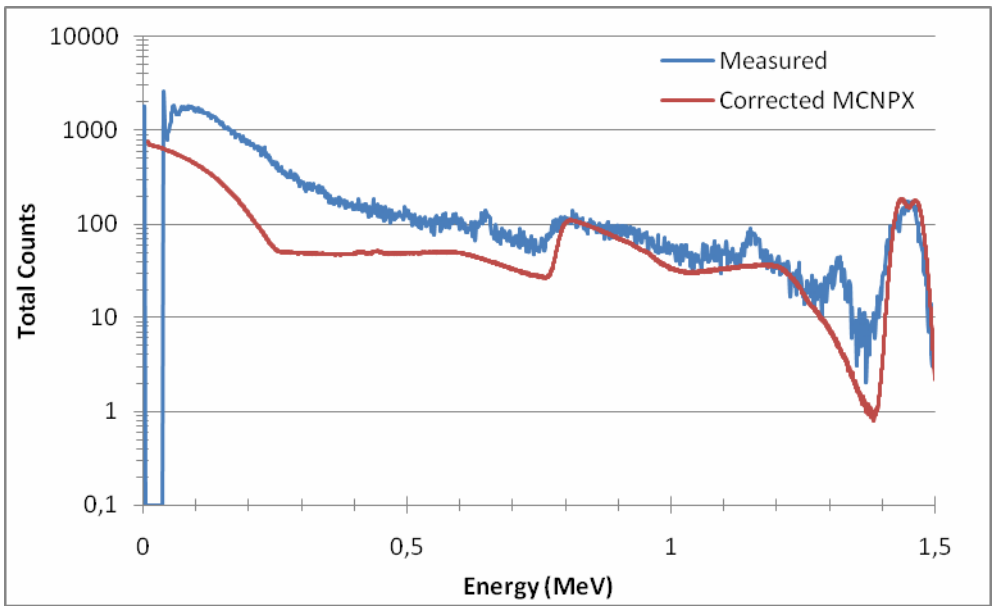
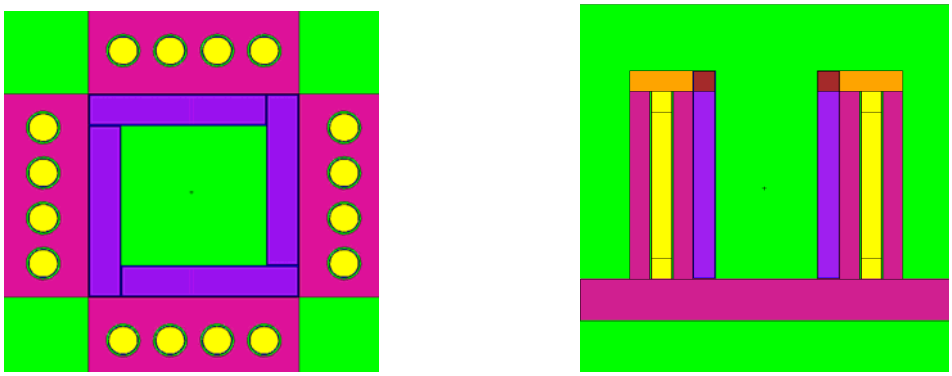


Figure 4. Measured and simulated for the $LaBr_3$ probe background spectrum

4. Simulations and Results

The detector concept consisted of 4 neutron detecting slabs and 4 gamma detection slabs. Each neutron detection slab contains 4 ^3He tubes of 2.54 cm OD with a 25.4 cm active length placed within a polyethylene slab[6]. Each gamma detection slab has 2.54 cm of scintillation material, a PMT placed on top, and a 1 mm aluminum casing. The gamma detection slab consisted of an Oleic acid matrix with 50% loading of $LaBr_3$ nanocrystals.[4] The detector concept is shown in Figure 5.



(A) (B)

Figure 5. Detector geometry (A) overhead view (B) cross-sectional view

The four samples listed in Table 1 were used to test the feasibility of this detector design concept. The samples consisted of a PuO_2 powder. Each sample had the same radius but had variations in fill height. The gamma and neutron simulations were executed separately.

Table 1. Detailed sample isotopic information

Sample ID	Pu Mass (g)	Pu238 (g)	Pu239 (g)	Pu240 (g)	Pu241 (g)	Pu242 (g)	Am241 (g)	O16 (g)
LAO-251	195.00	0.0975	142.1745	28.314	0.7995	0.6045	1.6575	23.01
LAO-252	365.00	0.1825	266.596	52.5965	1.46	1.1315	2.993	43.07
LAO-255	617.00	0.3085	450.41	89.0948	2.468	1.851	5.0594	72.806
LAO-256	436.00	0.218	318.498	62.9148	1.744	1.308	3.488	51.448

4.1 Gamma spectroscopy simulations

A separate deck was created for each isotope in photon mode. Each deck was executed with 1E8 particles, with 4000 energy bins, and a GEB function. Figure 6 is a plot of each individual isotope and Figure 7 is the sample plot imported to Canberra's Genie 2000[10] (Genie) gamma analysis software, both are for LAO-251.

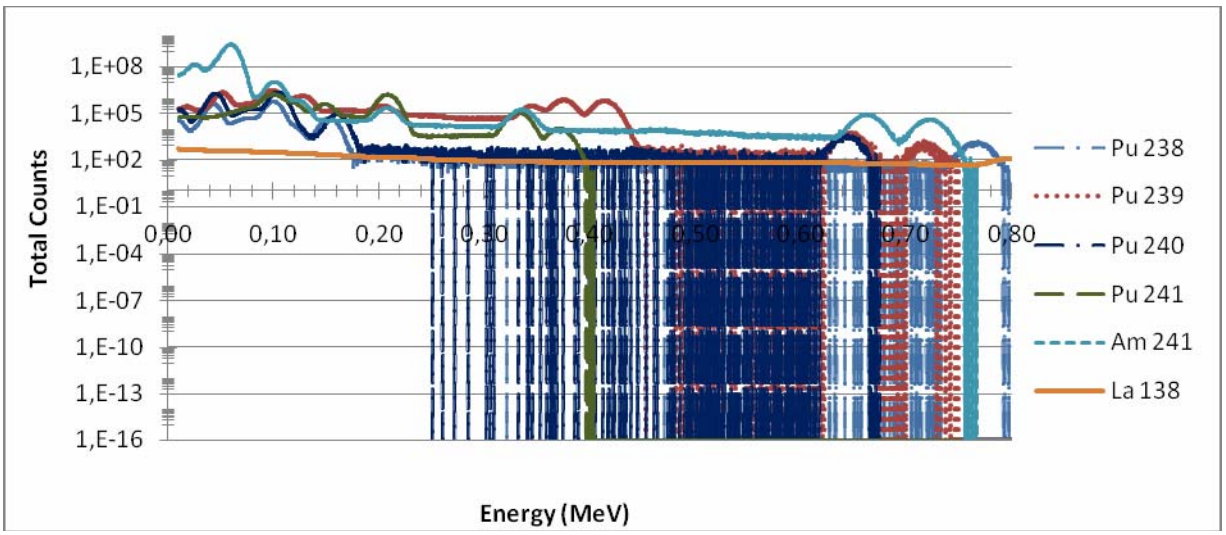


Figure 6. Counts vs. Energy for each individual isotope for $LaBr_3$ detection slab calculated with MCNPX

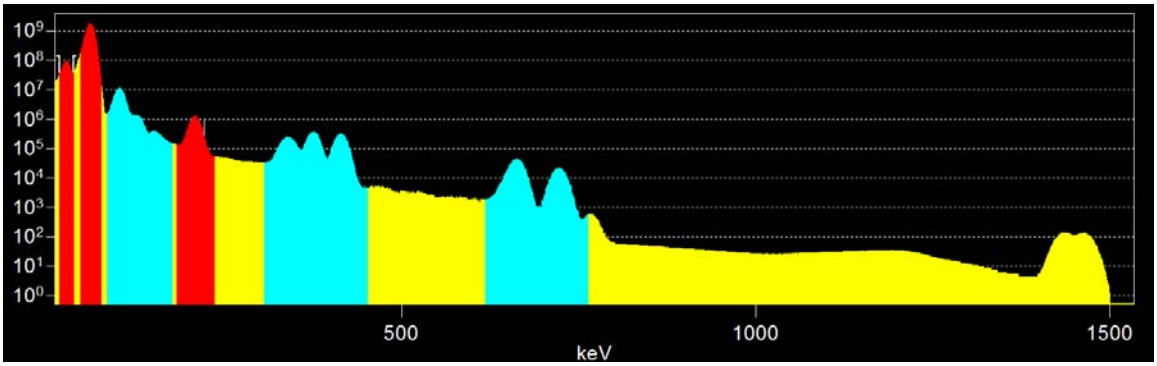


Figure 7. Imported LAO-251 spectrum in Genie 2000

From Figure 6, it can seen that all the isotopes are major contributors to the spectrum below 200 keV, but with the low resolution of $LaBr_3$ these mulitplet peaks are difficult to resolve. Because of this, no peaks in this area were used. A peak analysis was performed of the remaining peaks. It identifies the peaks and provides the net area counts above the continuum. This result provides seven with a range from 208 keV to 766 keV. The values from the efficiencies of the photopeaks can be seen in Table 2. Note that in Figure 7 the 766 keV peak is not highlighted, but was able to be used with Interactive Peak Fit, an analysis tool within Genie.

Table 2. Gamma detection efficiencies

Isotope	Energy	Efficiency		
	(keV)	LAO-251	LAO-252	LAO-255
Pu-241	208.00	10.87%	9.10%	8.01%
Pu-239	375.04	12.18%	12.54%	11.82%
Pu-239	413.17	11.42%	11.92%	11.43%
Am-241	662.42	9.72%	11.43%	11.16%
Am-241	721.99	8.51%	9.88%	9.75%
Pu-238	766.41	7.49%	8.31%	7.86%

4.2 Neutron simulations

For the neutron detector simulations, the same sample geometry was used with two decks for SF and (α.n), respectively. An MCNPX neutron capture tally was used. Each deck was executed for 1E7 histories. A plot of the singles count rate (left axis) and doubles count rate (right axis) versus $^{240}Pu_{eff}$ mass can be seen in Figure 8. In Figure 8 both lines are plotted but are virtually the same value. The associated error from the MCNPX simulation is less that 0.36%.

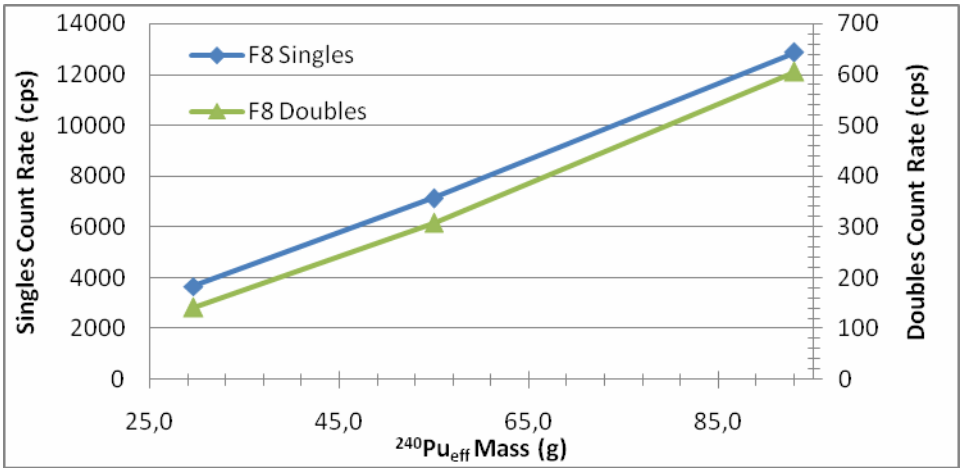


Figure 8. Count rate vs. $^{240}Pu_{eff}$ mass for singles and doubles

4.3 Mass determination

Using the data from both simulations and creating count rate vs. mass plots, similar to Figure 8, a linear relationship can be formed for each gamma peaks in Table 2 and $^{240}Pu_{eff}$ in Figure 8. This allows for equations to be produced that determine the mass of the isotope with only the raw count rate data, meaning no outside information about the sample was used. This provides values for all the masses in Equation 4 and 5 except ^{240}Pu and ^{242}Pu. ^{242}Pu is assumed to be zero based only a minute amount is produced within the sample through absorption. Note that the doubles information was used to calculate $^{240}Pu_{eff}$. This allows ^{240}Pu to be determined through Equation 4. The calculated masses are in Table 3.

Table 3. Calculated vs. Actual isotopic masses

Isotope	Calculated Mass (g)	Actual Mass (g)	Variation
Pu238	0.250	0.218	114.68%
Pu239	314.679	318.498	98.80%
Pu240	62.670	62.915	99.61%
Pu241	1.725	1.744	98.89%
Pu242	0.000	1.308	0.00%
Am241	3.534	3.488	101.33%
$^{240}Pu_{eff}$	63.300	65.612	96.48%

This in turn allows α to be determined which is used within the Neutron Point Model equations previously discussed and is shown in Table 4. Equations 1, 2, and 3 can now be solved as a system of equations.

Table 4. Calculated vs. Actual α value

	Calculated	Actual	Variation
α value	0.522	0.497	104.90%

The calculation yields an M of 1.08224 and a $^{240}Pu_{eff}$ of 60.78 g. The $^{240}Pu_{eff}$ is lower than both the

calculation and actual but was expected since M was included in the simulation of MCNPX, but since it

was just trying to be done this was disregarded. This shows that the isotopics can be determine with this experiment with little or no information about the sample.

5. Conclusion

Both detector systems are able to be put together and can benefit the other. Before, if a straight neutron count was taken it was required to calculate one of the three unknowns from alternate means, also little to no information is known on the odd isotopes. If just a gamma spectrum was taken it is unlikely that any information would be gathered from the even isotopes. When working together isotopics of the entire sample can be calculated and provide mass and identification of Pu. It could also be used in an environment for any type of nuclide identification. The gamma system has sufficient resolution to identify a number of isotopes, even if no neutrons are present.

There are some draw backs to this system that were unable to be accounted for. The resolution on the $LaBr_3$ slab was the same in this experiment as the probe. This information obtained does not include a FWHM value, and it was assumed to be the same as the probe, but may not always be the case. Also the total count for the gamma system is extremely high, and most gamma systems used today will not be able to handle a count rate of such magnitude. MCNPX can not include this in the model. This might require that the sample be placed farther away from the detector, which would also affect the neutron

count rate. Even with these negative aspects it is still believed that once the gamma detection medium is available the dual use detector system should be explored further and a prototype produced.

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SESSION 16

SECURITY IN INFORMATION TREATMENT AND EXCHANGE

NGAM, an advanced data collection instrument for IAEA Safeguards and general radiation monitoring applications

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Abstract:

This paper provides details of the design and operation of the Next Generation ADAM (NGAM), an instrument that has been jointly developed by Bot Engineering and the Canadian Safeguards Support Program as a successor for the widely deployed Bot Engineering ADAM unit. The NGAM hardware has been designed with a scalable architecture to accommodate evolving roles in IAEA safeguards and to serve as a base platform for various commercial applications. The instrument is capable of simultaneous collection of data from 8 independent nuclear detection channels and provides independent support of each detector, including independent bias generation. The NGAM is scheduled for delivery to the IAEA for evaluation during the summer of 2009.

Topics discussed in the paper include the enhanced web interface, scalability aspects, data security features, the unique data storage capabilities, chainable operation as well sample applications that the system has been targeted toward.

Keywords: radiation; monitoring; data collection; safeguards

1. Introduction

The radiation monitoring module currently used by the IAEA in the VIFM (VXI Integrated Fuel Monitoring) systems is now more than ten years old. While efforts have been made to extend the lifespan of the aging ADAM (Autonomous Data Acquisition Module) through the procurement of end of life components and increased maintenance, the ADAM is rapidly approaching end of life. In recognition of this fact, the Canadian Safeguards Support Program (CSSP) and the IAEA have been actively engaged in the specification and design of a successor for the ADAM, which is currently known as the Next Generation ADAM Module (NGAM). After detailed consideration and research, Bot Engineering has proposed that the original NGAM hardware requirements be upgraded to a version that will have both commercial and IAEA applications. In addition to commercially attractive features, the NGAM specification has been significantly upgraded beyond an exact functional ADAM replacement and in doing so has enabled the NGAM to fulfil the emerging IAEA requirements pertaining to remote monitoring and data access. This new device is now in final development and is expected to be commercially available in August 2009.

2. ADAM and VIFM history

While this paper is primarily written to describe the replacement unit for the ADAM (next generation ADAM module or NGAM), backward compatibility with existing VIFM systems is essential. The VXI Irradiated Fuel Monitor (VIFM) is a multi-channel system which uses multiple ADAMs to collect and analyze radiation data. In terms of system numbers, as well as number of detectors being monitored, it is currently the IAEA's widest deployed fixed radiation monitoring system and is a key element in the IAEA's OLR spent fuel monitoring implementation.

The original version of the VIFM was specifically designed to meet the IAEA's mid 1990's recommendation for all new instruments to conform to the IEEE VXI standards. The VXI recommendation was subsequently dropped by the IAEA in favour of the use of Ethernet interconnectivity. In response to the revised requirement, an Ethernet accessible version of ADAM was developed. The Ethernet Adam ("WebAdam") retained the original ADAM basic design but substituted an Ethernet interface and local power supply for the original VXI interface. The WebAdam was the first instrument to be provided to the IAEA which was compliant with their newly adopted Ethernet standard.

On a system level, the VIFM system was specifically designed to allow large data collection systems to be built and configured with minimal effort. This configuration capability included the ability to carry out data collection system configuration with simple on screen line drawings. This feature also incorporated the ability to install online analysis functionality inside of the data collection computer thus providing real-time signal analysis and safeguards relevant fuel material movement data. The rapid deployment capability combined with a standardized hardware and software platform has been proven to significantly decrease installation time and technical manpower cost.

The VIFM system was designed prior to the widespread adoption of remote monitoring; in particular, it had to support the needs of an inspector routinely visiting every 3 months to retrieve raw and summary data. It was later updated to accommodate remote monitoring. The NGAM, being a new design is expected to take the remote monitoring functions even further and address the deficiencies that exist in current remote monitoring implementations.

3. NGAM compliance with ADAM legacy characteristics

The ADAM is currently available in 2 variants, one VXI based and one Ethernet based. Aside from the digital data interface method, the module functions are identical and can in fact operate interchangeably under the VIFM system software. In general, the Ethernet based device has been used in portable systems as well as in fixed installations in which small numbers of detectors are needed. The VXI ADAM has been deployed extensively in applications where large numbers of detectors are needed, as illustrated by the fact that the VXI system is used routinely with 20+ detectors. These larger systems require that several ADAM's be connected into a data collection cluster in order to accommodate the full complement of detectors, something that is at times difficult with the WEBADAM due to its size. The NGAM module size has been significantly reduced relative to the WEBADAM so as to allow instrument clusters to be housed with a module density similar to that achievable with the single slot VXI form factor modules used by the VXI ADAM. With this reduced size and the VIFM collect software's ability to work seamlessly with both VXI and Ethernet interconnectivity, the NGAM can effectively replace VXI and Ethernet ADAM's over the entire span of system sizes.

The ADAM as well as the system detectors are designed for low power operation. The standard VIFM/ADAM system was designed to allow the ADAMs to operate for over 3 months on battery power alone. This requirement stems from the possible loss of AC power immediately after an inspection has occurred. Alternate measures have been selected to address such a scenario and as a result, the newer IAEA requirement is for 8 hours of battery operation. Never-the-less, with the newer electronic components that are currently available, the NGAM power consumption is similar to the original ADAM, even though the NGAM has significantly improved performance.

The original ADAM was designed to be able to accommodate different detector types with shaping times spanning a range of 30 nS to 5 µS range. The high speed capability has proven useful in high field radiation environments where large dynamic range without high field paralysis is important. This is a particularly important where irradiated fuel of various burn-up and cooling age can be encountered. An example of a real world requirement for the ADAM and NGAM is the monitoring of OLR discharged fuel. In this application fields in excess of 4 million Rads per hour are routinely expected to be quantified while, the same detector is also expected to detect long cooled fuel that emits gamma fields of only a few hundred mR/hr. This wide detection range places specific demands on the ADAM analog input sections, for which the NGAM must comply.

An ever present problem with modern electronics is the high rate of obsolescence. Both the ADAM modules and VIFM system suffer from this problem with many of the components having reached end of production life with more currently on end of life notification. While this problem is unavoidable, in any new design, certain steps can be taken to achieve optimum longevity. Many of these options were not available in the era in which the ADAM was designed, but can be used for the NGAM. These steps include the use of generic processors cores that allow the preservation of the software investment should the selected CPU's become obsolete, the use of military qualified components, which traditionally have very long availability times and the use of plug-in assemblies for high risk components such as memories.

4. NGAM Detector Interface

The ADAM supports a wide range of detectors and due to the installed infrastructure cost, these detectors must be able to be re-used in any system that is upgraded to use NGAM modules. The detector implementation used with the ADAM is a proprietary design that utilizes a single coaxial cable for the ADAM to detector connection. This feature is particularly important for monitoring in large facilities with long and difficult runs to detectors. Various styles of detector housings and electronics are in use ranging from waterproof designs to radiation hardened designs intended for use in spent fuel bays and for freshly discharged fuel measurements. Some detector assemblies have been qualified to over 10 MRad integrated gamma dose which are particularly useful in containment monitoring applications where the mean background dose rate can be quite high. There are also composite detectors used in a number of systems. These typically consist of combinations of detectors (gamma-neutron or gamma-beta for example) contained within a modular assembly. These assemblies are tamper resistant and are designed for rapid swap out in case of failure or radiation exposure induced end of life. The rapid swap out design is intended to minimize both MTTR as well as providing ALARA human dose during the swap out operation.

As noted, the NGAM must retain the ability to operate interchangeably with currently installed detectors, complete with compatible signal levels, timing, physical and mechanical interface. Above and beyond this legacy requirement, it is highly desirable to incorporate enhanced features into the NGAM that can allow various degrees of detector authentication to be carried out as an autonomous operation. Through the use a multi CPU with two DSP sections and a high-speed, high gate count FPGA, various detector authentication techniques, some of which are radiation tolerant to the mega RAD dose level, are possible with the NGAM architecture. This important capability is beyond the scope of the current discussion but will be covered in subsequent publications.

Figure 1: For small systems the Web Adam (left) is ideal but for large systems, the VXI system (right) can support 96 detectors

5. Miscellaneous requirements

The NGAM basic requirement for safeguards related roles is that of a multichannel nuclear radiation logger. The requirement included the need to be able to operate in manual stand alone applications and large computer controlled implementations involving many NGAM units. These operational modes have been accommodated in WEB ADAM via the use of a specialized HTTP interface. For legacy reasons this interface must be emulated in the NGAM. This approach has proven satisfactory in the past but is based on older TCP/IP standards which are now slowly being abandoned by the mainstream IT industry. For that reason, the NGAM supports the old WEBADAM operational modes and the newer HTTP and TCP/IP versions.

The ADAM was designed to utilize state of the art mass storage when it was designed. At that time, the best available memory consisted of linear array PCMCIA flash memory. Unfortunately, this memory technology was short lived and was quickly replaced by ATA compliant PCMCIA. In an attempt to avoid a similar experience with the NGAM, the mass storage system that has been selected is USB based ATA. The use of the ubiquitous USB standard in conjunction with the ATA mass storage standard removes the need for association with a physical memory standard. In basic terms, any USB communicating ATA compliant memory device can be used. At present the preferred device is the USB memory stick, which is expected to remain popular for many years. If however a new type of device emerges as long as it is supported by the USB physical interface, it can be used on the NGAM with field upgradable software.

Due to the ever expanding numbers of USB based peripherals, the NGAM is equipped with a total of 6 built in USB 2.0 host ports. While USB has the ability for simple expansion via low cost USB hubs, the on board USB ports allow quick, trouble free connection where power, addressing and controlled access is provided. Of the 6 ports, 2 are used for mass storage, 1 for a USB based authentication token and 1 for removable archive mass storage, should it be needed. The remaining 2 ports can be used for custom applications that may require functions such as GPS, WiFi, GPRS or even a camera connection.

6. NGAM in the non safeguards role

While the NGAM effort was initially aimed at the replacement for the ADAM and planned to be funded as such by the Canadian Safeguards Support Program (CSSP), it was quickly realized that there are significant advantages to increasing the scope of applicability of the NGAM to include other applications. An analysis of the possibility clearly indicated a reduction in life cycle cost will occur should a dual role be adopted. This is due to higher manufacturing numbers and increased fielded instrument operating hours and experience gained due to the diversity in operating conditions. Specially built equipment has typically required a long period of refinement in order to deal with problems that have arisen during use. With a larger more diverse deployment, any deficiencies can be more quickly identified and rectified. Similarly, the larger scale deployments that are possible with multi-use equipment allow the justification to increase the validation budget from a manufacturer's perspective. As a result of these factors, a unique arrangement has been implemented between private industry and the CSSP where the core IAEA NGAM functionality is funded by the CSSP and non-IAEA related enhanced features are funded by the commercial partner (Bot Engineering).

A number of additional uses for the NGAM are envisioned in governmental agencies, nuclear power plants and environmental monitoring. Consequently the NGAM specification was upgraded to an Enhanced NGAM (ENGAM) specification which included a set of new requirements. While not strictly required for the VIFM safeguards role, the capabilities will exist in each NGAM that is produced.
In developing these Non-Safeguards requirements, consideration has also been given to the possibility of expanded applicability within the IAEA. Fortunately, there is a certain degree of synergy between the possible IAEA roles and the commercial requirements, particularly in the future remote monitoring applications.

7. The enhanced NGAM joint use requirements

In order to satisfy requirements that have been identified as being needed in the commercial sector, the basic NGAM design requirements have been enhanced to include a number of new and novel features. After these features were designed into the instrument, we were pleasantly surprised to find that many fit into the concepts that the IAEA is developing for future use in the remote monitoring area. Some of the key specifications that fall into this category include:

- Ability to operate as a cluster without the need for a centralized collect computer
- Ability to provide real time analogue diagnostic information for remote fault diagnosis
- Ability for user to control device without need for a computer
- Provide capability for future expansion as a software development only

The above requirements have resulted in very specific hardware enhancements to the original NGAM concept. The resulting design, while being more complex, provides a greatly expanded instrument capability, which when combined with the wide range of available VIFM detector types provide a module which is capable of fulfilling any foreseeable safeguards role with minimal development effort.

8. NGAM Radiation tolerance

In the decade since the ADAM was originally designed, a large number of advances have taken place in the realm of micro-electronics. These include both digital and analog micro-electronics. These advances have both benefits and disadvantages in the nuclear instrumentation field. While it would seem that the NGAM design effort could only benefit from these advances, caution has been exercised in the design process. Of particular concern is the fact that many digital circuits have migrated to small geometries, with 0.18 µM or smaller being a relatively common fabrication scale. With this decrease in size has come cost savings for the chip manufacturer, higher levels of economical integration and unfortunately a much higher susceptibility to ionizing and linear energy transfer upset events. What once was only a problem with high speed SRAM's is now a problem with virtually every high density state retaining integrated circuit.

The problem of single event upset has been a well known and understood problem for at least 2 decades and a number of methods for mitigating this effect have been developed. Since the NGAM is intended to operate in a nuclear environment this aspect is tightly specified in the NGAM requirements and is dealt with using conventional techniques. It may be illustrative to note that the problem of radiation upset has been recognized for some time with the VIFM/ADAM system and the VIFM was up until a few years ago the only IAEA system which incorporated a radiation tolerance prior to design. This radiation specification has since been modified by the IAEA and has been adopted as a requirement for all new developments.

9. NGAM enhanced specification design characteristics

The ENGAM joint use requirements can be directly linked to several design features. These are summarized as follows:

- Ability to operate as a cluster without the need for a centralized collect computer

  This capability is provided by a TCP/IP, HTTP implementation that allows a single NGAM to act as a central control device. The implementation is such that failure of any controlling NGAM will result in the next NGAM in the network taking over its control role. This free form control architecture is augmented by the use of an SQL data base engine in each NGAM which can support both internal and external data retrieval requests. The combination of SQL and a floating control allows the complete network data to be archived on to a single mass storage device, which can consist of an ATA compliant hard or solid state disk. Data retrieval from the controlling module can be either locally or remotely carried out.

  As per the original NGAM requirement, collected data is tagged with an authentication field in each record group. This field becomes part of the SQL database and can optionally be examined as part of the data review process. One note regarding SQL is that SQL is based on the use of ASCII data. The NGAM accommodates this data form through either direct ASCII

data storage or in the case of the authentication field, modulo 16 conversion. In some applications such as list mode NCC, it is envisioned that modulo 128 conversions will be used for sparing of mass storage space.

- Ability to provide real time analogue diagnostic information for remote fault diagnosis

  In order to accommodate this requirement, the NGAM is equipped with the ability to capture and record the detector operating conditions such as detector current and analog output waveform. It also has the ability to control power to the detectors. The side effect of this capability is that the NGAM is equipped with a high speed ADC that when combined with a simple software upgrade, allows the NGAM to be used as a high performance DSP based MCA.

- Ability for user to control the NGAM without need for a computer

  The Original NGAM and WEBADAM were conceived as panel-less instruments. This requires that a user servicing the instrument use a computer to control operations such as safe mass archive storage removal. In order to avoid the need for an auxiliary computer, particularly in the case where there is no central collection computer, the enhanced NGAM specification was upgraded to include a small, interactive control panel that provides both instrument status readout to the user as well as a simple menu driven control system for setup and operation of the instrument. The control panel is based on a two button alpha numeric display panel that provides an interactive method of control. The messages, control functions and interaction language can be tailored to specific applications.

- Provide capability for future expansion as a software development only

  One of the goals of the enhanced requirement specifications was to develop a hardware platform that could be modified to add new or different functions as the need emerges. Ideally, these modifications would be possible without hardware design changes, and instead would be a totally software based exercise. To a large degree, this has been accommodated in the NGAM through the use of standardized software flow, a well specified parametrically adjustable data storage structure, the use of a very large FPGA and the incorporation of both analog and digital sensor interfaces. In most cases, instrument function can be modified through the reconfiguration of the front end FPGA. This is in essence a software exercise based on HDL coding. Once coded and validated, the FPGA can be updated via the FPGA JTAG interface that is connected to one of the NGAM processors.

  In most cases, the actual level of software development that would be needed for the addition of new data acquisition functions will be minimal since the NGAM software is designed to read configuration registers from the FPGA and adjust the data record size and sampling interval to match. Once data is packaged it is sent to the data base after which further access can be carried out via standard SQL query commands.

To accommodate the above listed capabilities, the NGAM utilizes 3 processors as a well as an FPGA based front end data buffer/acquisition controller. The architecture includes a floating point based DSP sensor interface section, a data storage and analysis section and a graphics and display intensive page server/user interface section. The multiple CPU approach is unique in that very high aggregate CPU throughput can be achieved with relatively low clock rates and very low power consumption. The multi-CPU approach also proves a high immunity to SEU events going unnoticed as the multiple CPU's constantly interact with each other and therefore must adhere to a strict error control protocol. In extremely critical tasks, it is even possible for two or more CPU's to execute mirror programs so that a result can be exchanged and compared.

Unquestionably the volume of software used in the NGAM is substantial when compared to earlier devices such as the WEBADAM. In order minimize the code development time and cost, a customized COTS small foot print real time operating system (RTOS) has been used as a means to easily support class level USB devices, the SQL data base server and web graphics interface. The source code

level code distribution RTOS that has been selected is well known in the industry and has wide scale of deployment in specialty equipment.

In terms of physical packaging, the ENGAM final form factor is currently in development but is near finalization with the prototype being shown in figure 2.

Figure 2: Prototype NGAM shown next to the WEBADAM

10. Future trends in connectivity and data security

The original VIFM system was designed during time when remote monitoring was not easily realised and IAEA routine inspections every three months were the norm. Since that time, remote monitoring is used in a large and ever growing number of facilities. Now that the approach for remote monitoring is well defined, the NGAM has been designed to accommodate the current and future needs of this data access method.

There are several features that would be highly desirable in future remote monitoring applications. These include:

- Real time remote access sensor and cable diagnostics

  While not strictly required for remote access, the capability is an important one in the event that a sensor failure occurs. A remote diagnostic capability provides information that is very useful in the site visit planning process since it is possible to estimate the level of effort that will be required for system repair. Appropriate spare parts and time can be allocated based on the remotely acquired diagnostic data. Real time access to the diagnostic functions is also important since many of the diagnostic steps previously carried out in the field can be now be done from the remote site.

  Diagnostics such as cable TDR, signal and noise level assessment, detector accumulated radiation damage measurement and electronic failures can be accommodated via the built in NGAM hardware once software is developed to carry out these functions.

- Elimination of a central collect and analysis computer

  This item has been mentioned as part of the enhanced NGAM specifications. It is listed here for further clarification. The use of a PC based central collect computer has been the standard approach in safeguards monitoring system for over 20 years. During this time the problems with this approach have become well known. These include reliability issues with both the PC hardware and OS, the need for periodic servicing of moving parts such as fans and disk drives and the difficulties in proving long term backup power in cases of prolonged AC power failure.

Elimination of the central collect computer is a rather simple exercise if only data collection and storage is needed. In cases where the safeguards system is required to carry out online, real time analysis, as is the case of many of the VIFM systems, the problem becomes much more complicated. The continued need for on-line analysis is not entirely clear, but it is certainly very convenient in cases of physical visits to the facility. In recognition that this capability desirable, initial work is being carried out to allow the NGAM to utilize its HTTP engine to be used as both a data review tool and a JAVA based analysis tool. Since the sensor data is stored in a SQL data base within the NGAM, analysis and review is greatly simplified given the ability to issue data recall queries that can locate periods of sensor activity very quickly. This ability to quickly focus on periods of interest makes it possible carry out run time analysis via the web browser interface, in a way similar to that currently available with the VIFM review tool.

- Low power consumption, Last mile connectivity

  Many applications are now emerging where safeguards must be applied to quasi remote locations such as dry storage container yards. Typically these sites have minimal to no available AC power and no Ethernet connectivity. In order to monitor these sites using remote access a means of powering and communicating with the NGAM system is needed. The NGAM uses a scalable power control system to adjust the level of power and performance the system exhibits based on the power source that is being used. In battery operation, front panel indications as well as enhanced web interface are switched to on demand operation, where they are in low power sleep mode most of the time. The power control also allows various USB devices to be powered off on command.

  Remote site wireless communications can be accommodated via the USB interface on the NGAM. GPRS, WiFi and proprietary communications standards are all available for USB based interconnection. If a particular device is selected, driver software can be developed rather quickly due the availability of class level drivers for the USB portion of the NGAM OS.

- Ability to accommodate changing authentication and encryption standards

  It is envisioned that over time, authentication and encryption standards will change. To deal with this and to simplify validation of the new implementations, the NGAM does not incorporate any internal security software. Instead, it is reliant upon the use of USB based FIPS certified security devices that are commercially available. Device specific software is needed for each device, but development is simplified as a result of the availability of class level drivers.

- Provision of a detector and cable authentication capability

  This aspect is very important and has been the focus of a new development for several months. With the high level of computing power available in the NGAM, a number of novel radiation hard authentication techniques are possible. These range from passive, one way fingerprint methods, to two way challenge based protocols. The NGAM one wire interface has been designed to allow multiplexing of several signals onto the single coaxial cable so that bi-directional, simultaneous digital and analogue signal transmission can take place without interference between signal streams. Several candidate authentication and identification methods have been identified and the final implementation could involve the use of several of these methods being combined into a single multi-layer approach.

- Guaranteed operation during manmade or natural power supply excursions

  In some operating environments, power reliability is very low and may be subject to variations that exceed those expected in a controlled environment. The NGAM must be able to accommodate such perturbations. In order to provide maximum flexibility in power supply

design and backup power, the NGAM operates from a wide range low voltage power source. Operation from 6 to 24 DC is possible for both the main and battery supplies. DC power as well as battery charge functions are accomplished via either standard or specially built external power supplies. In addition to the external power supply option, the NGAM uses a scalable Li-Ion battery with internal charge controller. This battery source is activated upon disconnect of external power sources and allows the NGAM to operate for a prescribed time period with no external power. This feature was originally incorporated to allow an orderly file system shutdown but can be scaled to operate the NGAM for extended periods of time if such a need emerges.

- High reliability data storage

  Data loss is a critical failure and must be avoided. The NGAM mitigates data loss through the use of triplicate data backup. Two copies of the SQL data base are stored, each on separate media and one copy of VIFM format compliant sequential time stamp data is stored on a separate archive media. The SQL and sequential file data paths within the NGAM are separate and in fact are controlled by separate CPU's. The use of triplicate data ensures that at least one copy of data is available in the event of CPU upset. In addition to this approach, 2 NGAM units may be powered from separate power sources in parallel to ensure continued system operation in the event of hardware failure.

- Secure method of remote tamper indication

  In a remote monitoring environment, equipment tamper events are not readily evident since physical inspection of the equipment does not take place during data retrieval. In recognition of this potential problem, the NGAM utilizes a multimode tamper indication system. This system depends on both individual physical sensors, and where available quorum sensing. The sensor detection system uses a hardware based write once register that records the time of the tamper event. This data is available as a discretely recallable variable available via a secure access protocol.

11. Current status

At present the NGAM design is complete and functional prototypes have been built. Testing has demonstrated compatibility with current VIFM data collect software. Six units are to be delivered to the IAEA for evaluation by the end of August 2009.

12. Summary

The NGAM is a follow-on development for the replacement of the current ADAM variants. The new device utilizes improved web interface and local control capabilities yet still retains interoperability with the original ADAM and VIFM systems. As a result of commercial interests, an elaboration of the original specification NGAM was carried out prior to the start of the design process. The initiative for this enhancement has been generated by the desire to provide an IAEA instrument that has wider applicability than just the IAEA alone so that a larger instrument fleet can be fielded. The result of this larger production will be the significant reduction of lifecycle cost. The enhanced hardware has other potential applications within the IAEA including new and more inclusive scope remote monitoring applications. It is hoped that the IAEA will recognize additional benefits and potential that result from these enhancements and consider the NGAM hardware as an option for other applications similar to what has been presented in this paper.

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Multiple Sources of Safeguards Related Data

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Abstract:
The international safeguards system has been built up over the last decades based on international agreements and voluntary offers from States aiming at demonstrating their commitment to non-proliferation. For the European Union and its Member States the set of commitments ranges from the safeguards agreements with the IAEA (INFCIRCs 193/263 and 290) and their respective Additional Protocols, through voluntary offers like INFCIRC 415, or reporting of Pu, Am and Neptunium, to data on incidents of illicit trafficking etc. Under the above, much safeguards related data is transmitted to the IAEA, in most cases via the European Commission.

This paper presents a review of safeguards data supplied to Euratom and the IAEA, their treatment in Luxembourg, and examines their complementarity and/or duplication. The question whether this plethora of information contributes to the strengthening of the safeguards system or it is an unnecessary complication is also addressed.

Note:
This paper refers to seven different IAEA Information Circulars (INFCIRCs), which have been summarised at the end of the paper in order to assist the reader.

Keywords: Safeguards data, duplication, Integration, European Union

1. Supranational Safeguards within the European Union

Two supranational systems of safeguards operate within the European Union,: Euratom Safeguards, whose legal basis is Chapter VII of the Euratom Treaty; and IAEA Safeguards, based upon agreements under international law between the member states of the EU, the Euratom Community and the IAEA. One of the tasks of the Euratom Safeguards System is to act as the Community System of Safeguards *vis à vis* the IAEA under these international agreements. Some EU Member States have chosen to complement the Euratom Safeguards System by national systems of safeguards, although these national systems have no direct role in the execution of IAEA safeguards.

The Euratom Treaty entered into force on 01/01/1958, The first accountancy declarations were received in 1959 and the first Euratom safeguards inspections were carried out in 1960 [1]. The provisions of Chapter VII (Safeguards) are complemented by an implementation Regulation which has been revised over the years to take account of the changing nature of safeguards. The current Regulation is usually known as Regulation 302/2005 [2]. This regulation is complemented by two Commission Recommendations, the first of which [3] provides guidance to operators on the changes which were introduced by the latest version of the Regulation, and the second of which [4] sets out a reference framework for high-quality nuclear material accountancy and control (NMAC) systems compliant with the legal requirements of Regulation 302/2005. Detailed safeguarding procedures for particular installations are set out in binding Commission Decisions addressed to the operators of the installations.
The legal basis for IAEA safeguards in the European Union is INFCIRC/193 in the case of the EU's non-nuclear weapons states, or INFCIRC/263 or INFCIRC/290 in the case of the UK and France respectively. The Euratom Community is a party to all three of these agreements. INFCIRC/193 is an agreement in implementation of Article III.1 and III.4 of the NPT, whereas INFCIRCs 263 and 290

represent voluntary offers by the UK and France to accept IAEA safeguards.

Following the advent of strengthened safeguards after the discovery of the clandestine Iraqi nuclear programme, each of these three INFCIRCs was completed by an Additional Protocol. Detailed provisions on implementation of these INFCIRCs are set out in the accompanying Subsidiary Arrangements, which are also binding under international law.

Thus, the main legal elements of supranational safeguards in the EU are Chapter VII of the Euratom Treaty and its implementing Regulation 302/2005, as well as INFCIRCs 193, 263 and 290 together with their respective Additional Protocols and Subsidiary Arrangements. These instruments are summarised in Table 1 below, along with their dates of entry into force.

This supranational legal safeguards framework generates very large flows of information between the various parties. For example, Euratom Safeguards receives around 2 million lines of accountancy declarations from operators per year.

In fact, the information exchanged is very varied in nature, going well beyond accountancy declarations. Operators must furnish comprehensive descriptions of the basic technical characteristics of their installations, and provide information on their annual activity programmes, and provide advance notification of imports and exports. Implementation of the Additional Protocol has added new categories of information, including site declarations, research and development activities, manufacturing of sensitive nuclear fuel cycle equipments, and transfers of equipment used in the civil nuclear fuel cycle.

In addition to these flows of information produced to comply with the legal requirements, there also exist some other flows of information, submitted or published on a voluntary basis, and concerning in some way safeguards in the European Union. The information may or may not be in the public domain. Some of the arrangements have been published as INFCIRCs whereas other information flows are made in response to requests from the IAEA's Board of Governors, the details of which remain outside the public domain.

Safeguards information specific to installations or locations is generally not in the public domain, because placing such information there could constitute a security risk, although data may be made publically available in an aggregated form for purposes such as annual reports.

Table 1. Principal Instruments of IAEA and Euratom Safeguards within the EU

Instrument	Content/notes	Entry into force
Treaty Establishing the European Energy Community	Chapter VII contains directly applicable provisions on Safeguards	01/01/1958
INFCIRC/193	(EU non-nuclear weapon states' NPT safeguards agreement)	21/02/1977
Protocol Additional to INFCIRC/193		30/04/2004
INFCIRC/263	UK Voluntary Safeguards Agreement	14/08/1978
Protocol Additional to INFCIRC/263		30/04/2004
INFCIRC/290	France Voluntary Safeguards Agreement	14-08-1978
Protocol Additional to INFCIRC/263		30/04/2004
Commission Regulation (Euratom) 302/2005 of 8 February 2005 on the Application of Euratom Safeguards	Replaced Regulation 3227/76 which in turn replaced Euratom Regulations 7 and 8	28/02/2005

2. INFCIRC/415 - Background

The most important of the voluntary arrangements for the provision of safeguards related information to the IAEA is INFCIRC/415 whose origins lie in the discovery of Iraq's clandestine nuclear programme. In 1992 the IAEA Board of Governors adopted a decision requesting its member states to voluntarily provide information [5] on certain safeguards related transfers which at the time, five years before the adoption of the Model Additional Protocol, were not required to be notified to the IAEA. This voluntary provision of information was known as the Voluntary Reporting Scheme.

In order to lead by example, the then twelve members of the EU requested the IAEA to publish the details of their offer to supply IAEA with information on production and transfers of pre-safeguards nuclear materials as well as with information on transfers of nuclear equipment and non-nuclear materials. This offer from the Twelve was published by the IAEA as INFCIRC/415.

In 1995 Austria, Finland, and Sweden joined the European Union, bringing the number of member states to 15. All three new member states addressed diplomatic notes-verbales to the IAEA aligning themselves with reporting under INFCIRC/415. Since then, a further twelve states have joined the European Union, but none of them have indicated that they wish to align themselves with INFCIRC/415 - which of course does not necessarily mean that they are not voluntarily reporting under the Voluntary Reporting Scheme. In any case, the IAEA does not name reporting states in the Safeguards implementation report and the Commission has no other means of knowing which, if any, of the new member states, continue to report under the Voluntary Reporting Scheme.

Beyond the fifteen EU member states which requested the IAEA to publish the details of their participation in the Voluntary Reporting Scheme no other state has done so, which again of course does exclude that other states are reporting under the Voluntary Reporting Scheme.

The part of INFCIRC/415 relating to declarations of uranium and thorium covers three areas:

1) imports and exports of pre-safeguards material
2) production of pre-safeguards material
3) information on nuclear materials (i.e. not pre-safeguards materials) gathered by Euratom Safeguards under the Safeguards Regulation, but not transmitted to the IAEA

Since its adoption, the Voluntary Reporting Scheme has evolved into an element of the Additional Protocol, but with some differences. It is therefore useful to compare the offer made under INFCIRC/415 with the requirements of the three principal safeguards agreements as well as with the requirements of their respective Additional Protocols.

3. INFCIRC/415 and Pre-Safeguards Materials

Pre-safeguards materials are defined by INFCIRC/193 Article 34 (c), which defines the starting point for the application of the safeguards procedures of INFCIRC/193 as material which is either produced in the Community or imported into the Community, and which is of a composition or purity suitable for fuel fabrication or isotopic enrichment. INFCIRCs 263 and 290 contain similar provisions.

INFCIRC/193 requires reporting of imports or exports of materials not having reached the stage defined in article 34 (c) (i.e. of pre-safeguards materials), although reporting is only applicable in respect of nuclear uses. INFCIRCs 263 and 290 do not contain equivalent provisions, however. INFCIRC/415 requires reporting on all imports and exports of pre-safeguards materials for all peaceful purposes.

Therefore INFCIRC/415 adds the following to reporting of import/export of pre-safeguards material according to INFCIRCs/193, 263 & 290:

1) Community imports/exports to/from UK and France
2) Community imports/exports of pre-safeguards material for non-nuclear purposes to/from an EU NNWS.

At a first glance INFCIRC/415 would seem to provide the IAEA with useful information which it would not otherwise receive, but this neglects the AP's article 2 a (vi), which requires declarations of imports/exports to/from the Community in respect of pre-safeguarded materials [6] (although for the UK and France only insofar as NNWS are involved). Therefore, it is difficult to see much justification

for continuing to report transfers of pre-safeguarded materials under INFCIRC/415.

The section of INFIRC/415 dealing with reporting of pre-safeguarded material also contains a commitment to report "total production", although rather confusingly "total production of …. nuclear grade purity" since it is difficult to understand how pre-safeguarded material could be of nuclear grade purity. In any case, it is difficult to see what useful information would not be covered nowadays by declarations made in accordance with the AP's Articles 2 a (v) and 2 a vi (a). (The UK and French APs provide the same information, but only insofar as NNWS are concerned).

4. INFCIRC/415 and safeguarded materials

As described in the previous section, materials subject to safeguards are defined by INFCIRC/193 Article 34 (c). (INFCIRCs 263 and 290 contain similar provisions).

In INFCIRC/415 the Commission agreed to provide the IAEA with information gathered under the Euratom Safeguards Regulation which is not required to be forwarded to the IAEA under the safeguards agreement. In practice, this means information gathered relating to nuclear materials in the two nuclear weapon states which is not reported to the IAEA – essentially nuclear material outside the facilities designated by these states for IAEA safeguards. In the case of France, the practice has been to not include information in this part of the INFCIRC/415 declaration which France has already submitted to the IAEA under INFCIRC/207.

By definition, the information provided under this part of INFCIRC/415 is not provided under the provisions of INFCIRC 263 or 290. Nor is it provided under the terms of the respective Additional Protocols. However, as it largely concerns the two nuclear weapons states, the added value to the IAEA of this information is questionable.

5. INFCIRC/415 and Equipment Transfers

The states adhering to INFCIRC/415 also committed themselves to report to the IAEA exports from the Community of the nuclear equipment and non-nuclear materials listed in Annex B to Part 1 of the Nuclear Suppliers Group Guidelines (INFCIRC/254 Part 1). Or rather, they committed themselves to report upon the issuing of export licences. It quite often happens that no actual exports take place against an issued export licence, e.g. because the licence was requested in response to a sales enquiry, which did not result in a sale. Furthermore, even if exports take place against a given export licence, the actual exports could take place some time later or be spread over a period. Therefore, in practice, the information on equipment exports gathered via INFCIRC/415 may not be of much use. In contrast, the Additional Protocol requires reporting of actual exports, including actual dates. Moreover, the Additional Protocol also provides for reporting of intra-Community transfers of nuclear equipment and non-nuclear materials.

Reporting is direct from the state to the IAEA. The Commission is not copied into the exchange and hence does not know which, if any, EU member states continue to report on equipment exports in accordance with either INFCIRC/415, or the Voluntary Reporting Scheme in general.

It is also worth noting a curiosity in this connection. Firstly, although the Euratom Community itself is an operator of nuclear installations, INFCIRC/415 makes no provision for the Community to report upon its own nuclear exports, in contrast to reporting practice under the Additional Protocol.

6. INFCIRC/415 Reviewed

The Voluntary Reporting Scheme was conceived as a stopgap measure to plug some of the holes in the IAEA safeguards system until the Additional Protocol was implemented. But, then implicitly once states had brought the AP into force, there would be no need to continue reporting under the Voluntary Reporting Scheme, and states would formally rescind their commitment, or in practice simply cease to report. In any case, it is difficult to discern the added value of the information submitted under INFCIRC/415 compared to the information submitted under the AP (apart from the information on nuclear materials in the weapons states not otherwise declared to IAEA).

The Voluntary Reporting Scheme, as its name suggests is purely voluntary. INFCIRC/415 makes this plain when it states that information

supplied will not be subject to routine verification. The safeguards agreements and their APs do provide the IAEA with verification rights in respect of submitted information. There is therefore a risk of creating legal uncertainty by reporting very similar information twice – once with verification rights and once without.

The Commission has reported on nuclear materials on behalf of fifteen EU Member States under INFCIRC/415 since it was published. The fact that INFCIRC/415 reporting is confined to fifteen out of twenty-seven member states is unsatisfactory. It is not known whether any of the twelve member states not reporting under INFCIRC/415 report under the Voluntary Reporting Scheme.

It would seem therefore that, given that the Community and all fifteen INFCIRC/415 states are reporting according to the AP, the most appropriate course of action would be to rescind INFCIRC/415. However, this would require the co-ordinated submission of Notes-verbales to the IAEA from the fifteen Member States concerned plus the Commission. Although, by no means impossible, some work would be entailed. This is an illustration of the dangers of making political commitments semi-binding by placing them in the public domain in such a way that it is difficult to retract the political commitment when circumstances change.

Of course, participation in the Voluntary Reporting Scheme by states which have not yet started reporting under the Additional Protocol provides the IAEA with useful information that it probably could not obtain by other means, so the above conclusions do not apply to these states.

7. INFCIRCs 207 and 549

INFCIRC/207 is a voluntary offer, dating from 1974, made by the three depositaries of the NPT (Soviet Union, United Kingdom, and the United States), in which they undertake to provide the IAEA with advance notification of their exports to non—nuclear weapons states of quantities of nuclear materials exceeding 1 effective kilogramme. Subsequently, France and China also indicated that they would apply the same procedures.

INFCIRCs 263 and 290 were both signed after the publication of INFCIRC/207. Even so only exports from facilities on the list of designated facilities are reported under INFCIRCs 263 and 290 to the IAEA. Therefore, the added value of INFCIRC/207 is that it includes exports (exceeding 1 kg effective) from non-designated installations.

INFCIRC/549 was published by the IAEA on request of Belgium, China, France, Germany, Japan, the Russian Federation, Switzerland, the United Kingdom, and the United States to indicate the common application by these states of a set of policies on the management and transfer of plutonium as set out in a document entitled "Guidelines for the Management of Plutonium", annexed to INFCIRC/549.

These Guidelines include a commitment by these states to publish annual statements of their holdings of all plutonium subject to the guidelines; and estimates of the plutonium contained in its holdings of spent civil reactor fuel, and indeed the annual statements are published by the IAEA in the form of addenda to INFCIRC/549, meaning that this information is in the public domain.

Four of the INFCIRC/549 states are member states of the EU. The Commission prepares the figures for one of these states, although the declaration itself is formally submitted to the IAEA for publication by the state itself. In fact, the information contained in INFCIRC/549 is destined for the international public in general rather than the IAEA, since the IAEA is already aware of the plutonium holdings of states with which it has comprehensive safeguards agreements.

8. Np/Am Voluntary Offer

Np and Am are not defined as fissile materials in either the Euratom Treaty or the IAEA Statute and hence there is no legal obligation on holders of these materials either to make declarations or to submit to verifications by Euratom or by the IAEA. Nonetheless, in the late 1990's the IAEA's Board of Governors requested IAEA member States on a voluntary basis to provide information on stocks, production and exports of separated neptunium and Americium, and where applicable to voluntarily submit facilities capable of separating Np to a verification regime the terms of which would be agreed by exchange of letters.

The Commission, in its own laboratories, carries out small scale research and development involving the partitioning of long-lived actinides in view of their

transmutation to nuclides with a shorter half life. By means of exchanges of letters with the IAEA in 2000 and 2002, the Commission undertook to make a voluntary declaration to the IAEA of its activities with, and holdings of Np and Am. Discussions were held in 2002 in order to agree the procedures for the exercise and the first verification exercise took place in 2003. Subsequently, the Commission has supplied the IAEA with an annual report on exports, holdings, and activities. During this time, only low throughput separation activities have been carried out, necessitating only a single verification visit per year, although throughput is so low that even a single visit per year is perhaps too much.

No particular problems have arisen during the time that the Commission has been reporting and submitting its facilities for verification. However, as a voluntary activity, it is important to maintain a separation from routine inspection activities. The information that the IAEA obtains from this activity is not directly obtainable by other means under the safeguards agreement and therefore this activity supplies added value and the Commission is pleased to be able to make a concrete contribution to strengthening the international non-proliferation regime.

9. Illicit Trafficking Database (ITDB)

All twenty-seven Member States of the EU participate in the IAEA's ITDB, as the three candidates for accession to the EU. As for the Commission, it participates in a limited way, and only receives the non-restricted parts of the incident reports.

The Commission does not supply any information to the ITDB on seizures which have been notified to the Commission through accountancy reports or other means such as advance information pending arrival of the seized materials in a material balance area. The reason for not informing the ITDB of information received through safeguards accountancy channels is quite simply that the Commission has no legal authority to share such information with the ITDB. Nonetheless, if the Commission observes that a member state has not submitted an ITDB report on a seizure, it will remind the member state concerned of the ITDB's existence.
Of course, once an accountancy declaration about a seizure has been made to Euratom, the information will be forwarded to IAEA safeguards in accordance with normal practices for transmission of accountancy data from the Euratom System of Accountancy and Control. However, in line with the deadlines in Regulation 302/2005, it may take up to four weeks for an accountancy declaration to arrive with Euratom. So typically, IAEA safeguards will in turn receive an accountancy declaration from Euratom six weeks after the original incident took place. In fact, information transmission can be even later if the seized material has not been promptly moved to an MBA, since reports are only submitted after arrival at an MBA.

The seizure of illicitly trafficked nuclear materials could be indicative of diversion, and therefore it is remarkable that the first official notification of the Commission could take place weeks after a seizure. Furthermore, this late reporting leaves the Commission in a delicate position with respect to IAEA safeguards, particularly if there has already been an ITDB report issued on the occurrence.

10. Conclusions

This paper has examined the main voluntary undertakings made by the European Union and its member states in relation to furnishing safeguards relevant information which complements the principal safeguards agreements and/or their Additional Protocols.

In the case of INFCIRC/415, however, it would be truer to say that the information furnished duplicates a lot of the information furnished under the Additional Protocol. As noted above, the continuing application of INFCIRC/415 poses a number of other difficulties. The Euratom Community and the IAEA should therefore carefully consider whether INFCIRC/415 should be rescinded.

However, for those states not yet applying the Additional Protocol participation in the Voluntary Reporting Scheme constitutes a useful half way house on the road to implementation of the Additional Protocol.

Although the information published under INFCIRC/549 by the non-nuclear weapons states may well appear to duplicate information provided to the IAEA under Comprehensive Safeguards Agreements, in fact INFCIRC/549 is intended to inform the international Community generally.

Some of the information supplied by the UK and France in accordance with INFCIRC/207

duplicates information which is supplied under the terms of INFCIRCs 263 and 290. However, this duplication should be viewed under the perspective of the political wish of the nuclear weapons states to be seen to be providing information beyond what they are required to do under the terms of their safeguards agreements with the IAEA.

The paper also shows that a firmer footing is required regarding reporting of seizures of seized nuclear materials to the Commission if the Commission is to be able to correctly discharge its own responsibilities regarding verification of non-diversion as well as be able to supply IAEA with timely accountancy declarations on seizures of nuclear materials.

It is important that the international Community avoids as far as possible providing safeguards information on a voluntary basis to the IAEA which is more or less the same as information provided on a mandatory basis, otherwise the result will be a loss of efficiency due to comparing and checking of slightly different information, or even legal uncertainty as to the IAEA's verification rights.

Summary of IAEA Information Circulars

INFCIRC/193, 14 September 1973
The Text of the Agreement between Belgium, Denmark, the Federal Republic of Germany, Ireland, Italy, Luxembourg, the Netherlands, the European Atomic Energy Community and the Agency in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons

INFCIRC/207, 26 July 1974
Notification to the Agency of Exports and Imports of Nuclear Material

INFCIRC/254, February 1978
Communication Received from Certain Member States Regarding Guidelines for the Export of Nuclear Material, Equipment or Technology
(The Nuclear Suppliers Group Guidelines. The current revision is INFCIRC/254/Rev.9/Part 1, dated 7 November 2007).

INFCIRC/263, October 1978
The Text of the Agreement of 6 September 1976 between the United Kingdom of Great Britain and Northern Ireland, the European Atomic Energy Community and the Agency in connection with the Treaty on the Non-Proliferation of Nuclear Weapons

INFCIRC/290, December 1981
The Text of the Agreement of 27 July 1978 between France, the European Atomic Energy Community and the International Atomic Energy
Agency for the Application of Safeguards in France

INFCIRC/415, December 1992
Communication received from the Member States of the European Community regarding the Provision of Certain Additional Information on Production, Inventories and International Transfers of Nuclear Material and on Exports of Certain Relevant Equipment and Non-Nuclear Material

INFCIRC/549, 16 March 1998
Communication received from Certain Member States concerning their Policies regarding the Management of Plutonium

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A note on Euratom safeguards information security

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Abstract:

The Euratom safeguards system is information-driven. It is designed to help the European Commission satisfy itself that common safeguard objectives are met across the EU. To implement this system, the Commission services collect data and evidence; carry out analyses and verification. This process entails interacting with various actors, public and private, within and outside the EU. Information is therefore handled in a controlled manner, to ensure its integrity and guarantee that any applicable confidentiality requirements are observed.

The legislative framework and recommended practices relevant to the security of information evolve slowly. In recent years this evolution has accelerated, partly in reaction to newly perceived threats such as terrorism but also responding to a general drive for more transparency in public governance.

This paper summarises the state of affairs about information security at EU level, having safeguards-related information in mind. The authors survey the applicable Community law and relevant key texts. Some of the aspects of information security involved are illustrated through reference to relevant ongoing projects, such as the remote transmission of safeguards-related data between industrial sites and the Commission headquarters.

Keywords: Euratom safeguards; safeguards-related information, information security, Community legislation.

1 About information and secrets

The words ‘secret’ and ‘confidential’ are understood in different ways by different people (cf. [12], [14], [13]). The fact that such words may have a meaning in common language but a different - prescribed - meaning in a security system is source of difficulties. As noted by Quist in [24], problems with the varying interpretations given to the marking ‘confidential’ were noted as early as 1907 as creating “considerable confusion and misunderstanding”. In spite of this, the word ‘confidential’ continues to be used today for security markings[2].

The first part of this paper is therefore be devoted to clarifying what ‘secret’ information means, and what ‘classified’ information means.

1.1 Secret information

Secret information can be distinguished from other categories of information by discussing the possibility of use and the possibility of access. Using this approach, Warusfel proposes in [11] four

[1] The views expressed in this document are entirely those of the Authors and do not engage or commit the European Commission in any way.
[2] For instance, in the UK Government’s protective marking system [31], or the IAEA’s new information security policy [32]

categories which can be roughly described as: ‘public’ information, ‘limited dissemination’ information, ‘information for private use’ and ‘secret information’. Table 1 presents these four categories.

‘Public’ information means, in this context, information to which access is unlimited and subsequent use is unrestricted. Typical examples include: most government web sites, published legislation, and works no longer subject to intellectual property rights.

‘Limited dissemination’ information means, in this context, information accessible only to a limited selection of people, not because it is protected, but simply because it is “hard to obtain”. Once obtained, however, users have no conditions on the use of the information. Typical examples include the so-called ‘grey literature’[3].

‘Information for private use’ means, in this context, information which is effectively protected, e.g. by intellectual property rights. Access to such information is in principle unlimited – although it may involve a cost –. However, even once in possession of such information, holders are not free to use it as they wish. Typical examples include patents, and also copyrighted works etc.

In the context of this categorisation, ‘secret’ information may then be defined as information subject to restrictions on both dissemination and use. Access to such information is limited (e.g. by access controls) and, moreover, its holders are not free to use it as they wish. This definition encompasses a wide array of ‘undisclosed’ information that includes: secret technical know-how, information provided ‘in confidence’, information subject to professional privilege and ‘classified’ information.

The reader may want to consult Errera et al. [22], who discuss in depth how these four categories of information can be understood in the context of the Euratom Treaty.

	UNLIMITED ACCESS	**LIMITED ACCESS**
FREE USE	PUBLIC INFORMATION, e.g. legislative acts, public domain works	LIMITED DISSEMINATION INFORMATION, e.g. ‘grey’ literature
USE SUBJECT TO RESTRICTIONS	PRIVATE USE INFORMATION, e.g. patents, copyrighted works	SECRET INFORMATION, e.g. trade secrets, medical files, ‘classified’ information

Table 1: Categories of information with respect to access and use

1.2 Classified information

The difference between ‘classified’ information and other forms of information subject to secrecy is the use of an organised security system and the possibility of criminal penalties for security breaches.

Early on, governments understood the need to organise the protection of military secrets using some form of markings. According to Quist [24], the first organised use of security classification markings by a governmental entity is said to be by the British War Office during the Crimean War (1853-1856). Documents for use by the Cabinet were then marked “Confidential”, “Private Confidential”, or “Secret and Confidential”. At that time, however, the system still lacked many elements of a modern security regime such as procedures for clearance, document handling, etc.

Initially reserved to military secrets, security classification systems have grown to cover many other areas. According to Quist [24], already in the XVIII century in the United States, Congress had established secret committees to deal with e.g. correspondence with allied nations. As noted by

[3] Grey literature is best defined as literature which cannot readily be acquired through normal bookselling channels or libraries and which is therefore difficult to identify and obtain. (see ref. [1] for further information). For instance in the late 1950s and 1960s, detailed reports on the Eurochemic reprocessing plant were circulated amongst fuel reprocessing experts; these unclassified reports are today very difficult to find and obtain.

Donnely in [30], defense information and diplomatic negotiations are two of the six so-called “real secrets” in government[4].

Secrets, i.e. information subject to restrictions on disclosure and use can be categorised according to two dimensions: whether the protection is organised as a formal security system and whether breaches can be the object of criminal prosecution. Four categories can therefore be enumerated: plain secrets, privileged information, proprietary information and classified information. These categories are illustrated in Table 2.

‘Plain secrets’ is secret information subject to de facto protective measures which are, however, not necessarily organised as a security system. Such information will normally not be protected by law. A typical example would be an encrypted file with information about business customers, or a peculiar and secret technique for machining a material.

‘Privileged’ information is secret information protected by a legal privilege, such as information exchanged between a client and his attorney, or information received by a government official in the course of his duties. Such information, although it must be protected, is normally not subject to an organised system of security.

‘Proprietary’ information (also ‘undisclosed’ information or sometimes ‘protected’ information) means information which is protected by an organised security system, although not necessarily protected by law. Such a situation is common in companies such as industrial businesses. Frequently such a security system is organised as part of the company’s quality management system and includes features such as: measures for the physical protection of marked documents, handling procedures and disciplinary sanctions, etc. However, security failures in this context may normally not be the object of criminal prosecution. Nevertheless, they may lead to e.g. litigations for breach of contractual confidentiality obligations.

‘Classified’ information is, according to this categorisation, characterised by the fact that information is subject to an organised security system which is recognised by law. Breaches of security may therefore lead to criminal prosecution, which does not preclude disciplinary sanctions and/or contractual liability.

The reader will note that, in practice, the expression “classified information” is frequently misused in a corporate security context to mean e.g. “proprietary information”.

	PROTECTED DE FACTO	**PROTECTED BY LAW**
NOT SUBJECT TO A SECURITY SYSTEM	PLAIN SECRETS, e.g. business secrets, secret technical know-how	PRIVILEGED INFORMATION, e.g. personal data
SUBJECT TO A SECURITY SYSTEM	PROPRIETARY INFORMATION, e.g. trade secrets, information, covered by a non-disclosure agreement	CLASSIFIED INFORMATION, e.g. defence secrets, official secrets etc.

Table 2: Types of secret information with respect to protection and control

1.3 Security regimes

The common feature to ‘proprietary’ information and ‘classified’ information is that both are protected by an organised security system. The difference is that the latter system is formally recognised by law. This distinction can be source of misunderstandings since, for many security systems, an infringement may lead to action before a court (e.g. litigation for breach of contractual confidentiality clauses) even if the system itself is not the object of a statutory instrument.

[4] The list of “real secrets was proposed in 1974 by the National Security Advisor to US president Kennedy, McGeorge Bundy. The other four are: information about covert foreign activities, information about covert collection of intelligence, legitimate secrets about presidential decision processes and material with “capacity for international embarrassment” – e.g. assessments about foreign leaders.

It is therefore useful to discuss how security systems can be categorised with respect to the degree of legal protection offered, i.e. what type of responsibility is engaged. Table 3 summarises the situation for a number of security systems relevant to safeguards.

Infringing a security system can engage three different types of responsibility: disciplinary, civil and penal. The possibility of having recourse to disciplinary action is a basic feature of any mature security system. In most legal systems, the breach of a corporate security system can give rise to civil action, e.g. against an employee, or against a contractor. But only a handful of security systems are recognised in national law, e.g. state secrets, defence secrets.

In the context of national law, characterising a security system poses little difficulties. For a security regime to contemplate the possibility of criminal penalties, it must necessarily refer to the authority of the State, or to that of an entity exercising State powers.

The situation is more complex in the context of international law, since the concept of criminal penalties does not have a natural meaning. The security regimes of international organisations must therefore be examined on a case-by-case basis. Most do not open any possibility for judicial remedy. The security regime of the Organisation for the Prohibition of Chemical Weapons (OPCW) typifies this fact: the possibility of judicial pursuits against a breach of confidentiality is explicitly excluded. Another example is afforded by the security regime of the IAEA, since only disciplinary measures will normally be possible in case of a security breach.

The security regimes in force in the European Communities are in this regard unique. For instance, the Euratom Treaty explicitly provides in its Art 194 EA for the possibility of criminal sanctions by each Member State in the event of a security breach.

TYPE OF SECURITY REGIME	RESPONSIBILITIES		
	PENAL	CIVIL	DISCI-PLINARY
SENSITIVE BUT UNCLASSIFIED[5]	×	○	●
CORPORATE CONFIDENTIALITY [6]	×	●	●
NATIONAL SECURITY CLASSIFICATION[7]	●	●	●
EURATOM CLASSIFIED INFORMATION (ECI)	●	●	●
EUROPEAN UNION CLASSIFIED INFORMATION (EUCI)	○	●	●
INTERNATIONAL ORGANISATION CONFIDENTIALITY[8]	×	○	●

Table 3: Consequences of security breaches
(● = action possible; ○ = sometimes possible; × = normally not possible)

2 Security regimes in the Community

In the European Union, there are at least twenty-nine security systems recognised in law, i.e. security classification regimes. There are twenty-seven State (or defence) security classification systems. And there are two Community systems: EU classified information (EUCI) and Euratom classified

[5] Sample markings include 'CONFIDENTIEL', 'DIFFUSION LIMITEE' (used by the French authorities), 'COMMERCIAL-IN-CONFIDENCE' (used by British authorities), etc.
[6] Sample markings include 'BNFL IN CONFIDENCE' (used by BNFL), etc.
[7] Sample key acts include le "Décret 'secret défense'" (in French law), the "Official secrets act" (in British law), the "Sicherheitsüberprüfungsgesetz" (in German law), etc.
[8] Sample markings include 'SAFEGUARDS CONFIDENTIAL' or 'HIGHLY CONFIDENTIAL' (used by the IAEA), 'OPCW CONFIDENTIAL' (used by the OPCW), etc.

information (ECI). A detailed analysis of the full landscape of security systems in the EU is beyond the scope of this paper.

The existence of two security systems recognised in Community law can be source of some confusion and the following sections are devoted to clarifying some differences between the two. The key message is that the EUCI and ECI regimes are quite different and to a large extent complementary. They serve different purposes and do not overlap.

2.1 Transparency and access to information

Before delving into how the E(U)CI security regimes are implemented, it is necessary to mention the topic of transparency. Transparency is a fundamental principle of the European Union, enshrined in Article 255 of the Treaty establishing the European Community which provides that any citizen of the Union, "shall have a right of access to European Parliament, Council and Commission documents, subject to the principles and the conditions to be defined".

According to the Treaty, the institutions are required to fix "general principles and limits on grounds of public or private interest governing this right of access to documents". In other words, the fundamental principle is transparency and protection of information by secrecy must be justified. This is why, in 2001, Regulation № 1049/2001 was adopted to establish the rules regarding public access to European Parliament, Council and Commission documents [9].

Due to the limited space, this topic of transparency is not further discussed in this paper. Also not discussed are the important questions about secrecy and transparency raised in connection with the Aahrus convention on the access to environmental information.

2.2 Security regimes relevant to safeguards in the Community

The EUCI regime is established using different legal bases for the Commission and for the Council. In both cases the EUCI system is laid down as part of the Rules of procedure of the concerned institution. For the Commission, it was adopted by the Commission Decision 2001/844 on 29 November 2001 [5]. For the Council, it was adopted by the Council Decision 2001/264 on 19 March 2001 [6]. These two decisions have created security systems which are essentially identical. In a letter of understanding, the Commission and the Council have engaged themselves to keep the two security systems aligned. It is therefore reasonable to speak of a single EUCI security regime.

The ECI regime has a completely different nature. The existence of the ECI security system is provided for in primary law, namely in Article 24 of the Euratom Treaty, which reads "Information which the Community acquires as a result of carrying out its research programme, and the disclosure of which is liable to harm the defence interests of one or more Member States, shall be subject to a security system in accordance with the following provisions […]". The ECI security system is further implemented by a Regulation №3 adopted on 31 July 1958 by the Council [4].

The ECI security system must be read in conjunction with Article 194 of the Euratom Treaty. In simple terms, Article 194 creates an obligation of "professional secrecy", i.e. a permanent obligation on whoever is given classified information in relation with the Treaty to keep such information secret. Although the text of Article 194 does not explicitly refer to 'classified' information, it implicitly does so since it covers "any facts, information, knowledge, documents or objects which are subject to a security system in accordance with provisions laid down by a Member State or by an institution of the Community". Importantly, the duty of professional secrecy established by Article 194 provides for prosecutions according to national criminal law in the event of security breaches [33].

The two regimes, ECI and EUCI, are very different from a legal perspective. This has consequences in terms of the possibility for sanctions and for direct effect – e.g. the possibility of invoking the security regime in front of national courts –. The EUCI system is established by secondary law and ultimately draws its basis on the basis of the principle of institutional autonomy. The provisions establishing the EUCI security system are only a part of the "Rules of procedure" of the institutions. This limits the possibility of direct effect and limits the scope of potential sanctions. The case for ECI is very different. The ECI regime is established directly in the Treaty and is implemented by a Regulation which

explicitly provides for direct effect in all Member States. With respect to sanctions, the ECI regime is much more stringent, since Article 194 opens the possibility for criminal prosecution.

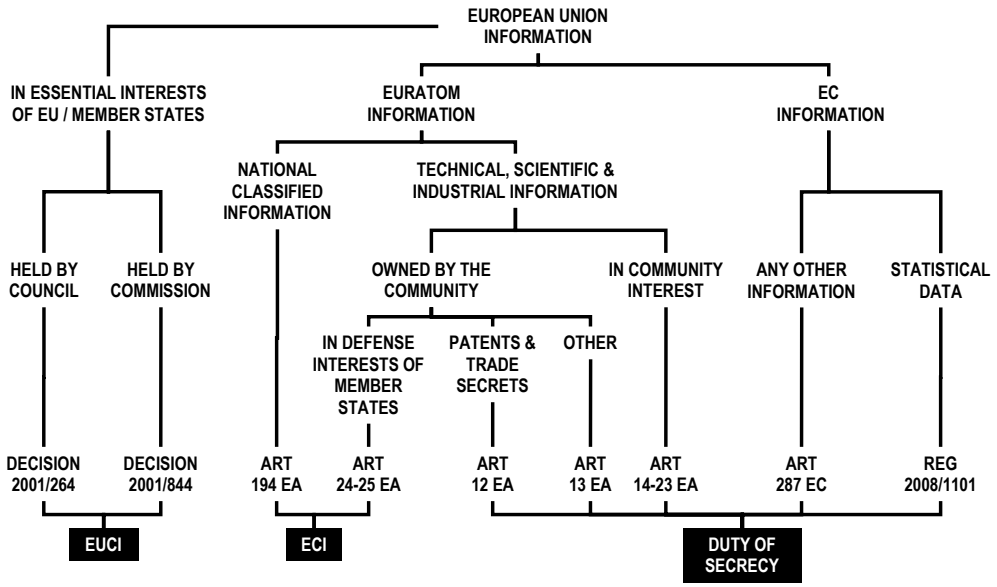

Figure 1: Community security systems with respect to the interests protected

2.3 Organisation of security in the Community

Figure 2 presents an overview of the architecture for implementing the E(U)CI security regimes in the context of the European Union. Not shown in the left branch of the figure, there are security-specific roles assigned to certain posts within the institutions[9], including various functions[10] important for IT security. Not shown in the right branch of the figure, there are detailed implementation documents[11] that supplement the legal basis, including documents derived from international security standards[12].

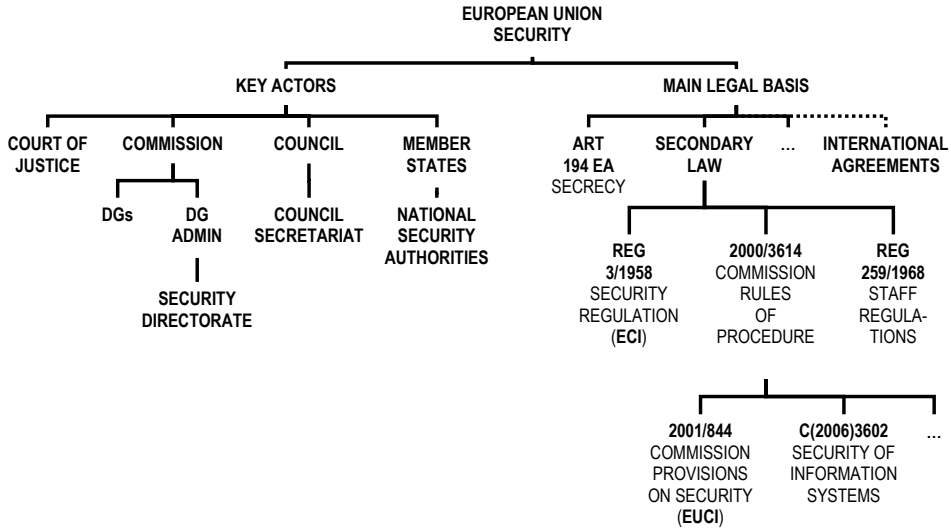

Figure 2: Overview of the organisation of security in the European Union

[9] E.g.: central and local registry control officers, security officers, IT security officers, system security officers etc.
[10] E.g.: project owners, system suppliers, system managers, etc.
[11] E.g.: classification guides, security notices, implementing rules, policies and guidelines
[12] E.g.: ISO/IEC 27001, PKI, Common Criteria, etc.

2.4 Information security in relations between the Community and third parties

The Euratom Treaty establishes the exclusive competence of the Commission in organising the dissemination of information related to nuclear matters. This competence also covers external relations. The Treaty provides that only the Commission shall be competent for "concluding any agreement or contract for the exchange of scientific or industrial information in the nuclear field" whenever "the signature of a State acting in its sovereign capacity" is involved.

This competence of the Commission extends to all information "which is of use to the [Euratom] Community in the attainment of its objectives", including information that "has been acquired subject to restrictions on its use or dissemination". In particular, this means that the Commission has a general exclusive competence for handling classified information in relation to the Treaty, i.e. "any facts, information, knowledge, documents or objects which are subject to a security system in accordance with provisions laid down by a Member State or by an institution of the Community".

The situation is different, however, where the concerned information is liable to harm defence interests of one or more Member states. Such information then falls under the scope of articles 24 and 25 of the Treaty and must be subject to the security system governed by Regulation № 3/1958 of the Council. In such cases, the competence of the Commission must be shared with the Member States.

In the case of safeguards-related information, international agreements play an important role. Any exchange of nuclear materials with third parties will normally involve the secure exchange of protected information, quite often classified. The conditions for such exchanges are normally laid down in cooperation agreements. In the case of Euratom safeguards, the two main types of agreements to consider are the agreements with the IAEA and the agreements with third States.

Regarding the IAEA, the exchange of information is, in principle, governed by an international agreement concluded in 1975 between the Euratom Community and the IAEA. The provisions for exchange of information in this agreement may be supplemented by annexes governing the exchange of particular categories of information, although this possibility has been to date little used. Further provisions concerning the exchange of information are given in the agreements concluded with the IAEA in respect of the application of safeguards in the European Union.

Regarding third States, the exchange of information is governed in most cases by provisions agreements for the cooperation.

In detail, the following international agreements contain specific provisions affecting the exchange of safeguards-related information:

- The agreements done with the IAEA for the application of safeguards in connection with the NPT, concluded respectively in 1973, 1976 and 1978, along with the corresponding Additional Protocols, all concluded in 1998
- The Agreements for cooperation on the peaceful uses of nucelar energy concluded with: the USA in 1995, Canada in 1959, Australia in 1981, Argentina in 1997, Uzbekistan in 2003, Ukraine in 2005 and Japan and Kazakhstan in 2006; and the Partnership and cooperation agreement concluded with Russia in 1994.

In principle, the exchange of classified information with third States or international organisations is only possible on the basis of a suitable between the European Union and the concerned third party. In this respect, the situation regarding EUCI and ECI is somewhat different.

For ECI, Article 194 of the Euratom Treaty implies that ECI security rules "shall not prevent application of special provisions resulting from agreements concluded between a Member State and a third State or an international organization". Moreover, Article 34 of the Regulation № 3/1958 defining the ECI security system provides that ECI security rules may not be opposed "to the obligations of the Community and the Member States (or of the Community or the Member States) in this field arising out of treaties or agreements concluded with third countries, an international organisation or a national of a third country". In other words, where the exchange of specific information is provided for in international agreements, the mere fact that the requested information is classified does not justify a refusal to transmit it. Either way, the Commission must make sure, prior to any transmission, that both

the transmission and the subsequent handling of the information comply with the security rules applicable in the Community for the category of information in question.

For EUCI, the transmission of classified information is conditional on the existence of suitable agreements between the parties. At the time of writing, the European Union had concluded security agreements covering the exchange of EU classified information with seven states (Norway, FYROM, Ukraine, Iceland, USA, Croatia, Switzerland) and only three international organisations (NATO, ICC, and ESA).

3 Safeguards information security regime

Given that, at community level, there exist different security regimes, the question may be asked about which regime to apply in the case of Euratom safeguards (chapter VII of the Euratom Treaty).

In practice, the ECI regime has been applied since its origin in 1958. This solution is clearly satisfactory since it affords the best protection for the secrets entrusted to the Commission in relation with its safeguards activities. From a legal perspective, this solution does raise some questions since, in principle the ECI regime is meant to apply only for a narrow category of information.

3.1 Historical background

As early as 1959, the need to apply safeguards to installations handling special nuclear material that was subject to particular security rules triggered a debate in the Council. Indeed, to give effect to the provisions in Article 79 of the Euratom Treaty, the Euratom Commission had adopted on 12 March 1959 Regulation №8 to define the nature and the extent of the requirements for accounting and operational records referred to in Article 79 of the Treaty.

This Regulation, which was to enter into force on 1 June 1959 provided inter alia that “Producers and users of source materials or of special fissile materials shal, by the fifteenth of each month at the latest, communicate to the Commission in respect of each of their installations: (a) A monthly return […] of materials in hand during the preceding month, together with a summary of the dispatches and receipts of materials during that month, showing in respect of each : date, quantity, composition, form and supplier or consignee; (b) A statement (hereinafter called "inventory") of materials in hand on the last day of the preceding month.”.

At the time, France observed that some of the information whose communication was required by the Regulation was protected according to French law as national defence secrets. Whilst the French authorities confirmed their willingness to abide by the Regulation № 8, they required the Commission to offer sufficient guarantees that the information would be subject to an adequate security system.

In reply to the issues raised by Frence, the Commission turned to the Council Regulation №3, adopted one year earlier on 31 July 1958 (see ref. [4]), establishing a security system for the protection of Euratom Classified Information (ECI). That Regulation, which gave effect to the provisions of Articles 24 and 25 EA had been designed to protect the scientific, technical and industrial information the unauthorised disclosure of which could pose a threat to defence interets of Member States.

In The Commission representative then informed the Council that the Commission had decided to extend the application of the ECI regime provided for in Euratom Council Regulation №3/1958 beyond the scope of Articles 24 and 25 EA to also cover the information being communicated under secrecy conditions in the name of the Euratom Commission Regulation №3/1959. In particular all information related with Article 79 EA declarations became from that point on subject to the ECI regime. This has not changed since.

3.2 Present situation

Today, the security of safeguards information finds its basis in a combination of the provisions of the ECI regime, of the provisions related to Article 79 EA, and of the commitment of the Commission

expressed in 1959. A detailed analysis of the resulting reference system of applicable texts is beyond the scope of this paper. The following texts form a core of that reference system:

- Euratom Treaty, particularly articles 12-29, 77-84 and 194 [2]
- Regulation № 302/2005 (Application of Euratom safeguards), particularly articles 20, 21, 34 and 35 [3]
- Regulation № 3/58 (Security regulation implementing Art. 24 EA) [4]
- Decision 2001/844 (Commission provisions on security) [5]
- Council Decision 2001/264 (Council's security regulations) [6]
- Decision C(2006)3602 (Security of information systems) [7]
- Regulation № 259/68 (Staff regulations of EU officials) [8]
- Regulation № 1049/2001 (Public access to documents) [9]
- Regulation № 354/1983 (Historical archives) [10]

Within the Commission, the reference system is completed by implementation guidance which is for the most part based on the relevant International Standards (ISO/IEC 27001, CC, PKI, etc.).

The security of safeguards-relevant information takes on particular importance in a number of contexts:

- Where the information may conduct to reveal critical commercial or industrial secrets;
- Where the information may concern the physical security of nuclear materials, e.g. revealing security vulnerabilities;
- In the context of the use for peaceful purposes of special nuclear materials formerly assigned to defence purposes;

The following examples illustrate some typical categories of safeguards-relevant information eligible for security classification, according to the current security rules:

- Information on security measures applied on materials and installations
- Safeguards-related data collected at the installations, incl. remotely transmitted data
- Advanced notifications of special nuclear materials transfers
- Safeguards-relevant information possibly revealing national defence / state secrets
- Information about proliferation-sensitive technology, e.g. ultra-centrifuge technology

In practice, most of the protected information held by the Commission services has been protectively marked by the Member States.

4 Illustrative cases

4.1 Remote transmission of data

The idea of using electronic networks to remotely access safeguards-related data and thereby achieve more effective and efficient controls is more than 30 years old. In 1978, the RECOVER (REmote COntinuous VERification) project was proposed [34] to test the concept of transmitting information on the status of C/S equipment directly to the Vienna Agency using telephone lines. At the time, the concept found difficulties in gaining acceptance as it was not found to be cost-effective, partly because only status-of-health information was sent, rather than the actual safeguards data.

In the late 1970s and early 1980s, the various programs for development of fast breeder reactors reached peak activity. Many experimental reactors were operating and various commercial size prototype were envisaged[13], e.g. Phénix, Superphénix in France, SNR-300 in Germany, PFR in the UK. Against this backdrop, most analysts were anticipating that by the end of the century, breeders would enter more and more into the picture. This implied not only that more reactors would need to be safeguarded, but as Schleicher and Miranda [35] concluded in 1981 that “more and bigger fuel fabrication and reprocessing plants with more complicated techniques would have to be placed under

[13] Elsewhere around the world: MONJU in Japan, CRBR in the USA, and BN-350, BN-600 in the USSR

safeguards" and that "safeguards authorities as well as operators have to be prepared for this development". According to Schleicher and Miranda, taking into account of safeguards requirements at the design stage "is of the utmost importance". Recalling the RECOVER program, they went on to consider that "the new possibilities of data transmission over long distances might furthermore reduce inspection efforts, by avoiding time and cost-intensive travelling by means of interrogation from headquarters."

By 1995, the IAEA was conducting tests in Sweden, Finland and Hungary in view of letting state operators change videotapes and sending them back to the IAEA, with special techniques being used to prevent tampering or substitution. At the same time, the US was conducting in Austraila tests of a remote-monitoring concept based on cameras and satellite links for real-time C/S. [36]

In 1999-2000, joint work between Euratom, IAEA, German authorities and a German operator was underway to prove the technical feasibility and reliability of remote data transmission. To this end a joint field trial was conducted at the Ahaus spent fuel storage facility, using a digital image surveillance system with three camera units. Starting in May 1999 and ending in July 2000, the field trial collected digital surveillance data and successfully transmitted it via ISDN to Euratom headquarters in Luxembourg for reviewing and archiving. The participating organisations concluded that, from the technical point of view, the field trial of remote transmission of digital image data was successful, but that cost aspects required some more consideration.

The rapid development of technologies and standards (ADSL, VPN) in the mid 1990s and subsequent widespread availability of secure channels over broadband networks at low cost from 2000 onwards made possible further progress in the development of remote data transmission solutions for safeguards data. This is illustrated by the case of a secure remote data transmission link from the Sellafield site to the Euratom safeguards headquarters in Luxembourg.

4.2 The secure link with Sellafield

The Sellafield site is one of the largest nuclear sites in the world. The site is home to key installations in the UK nuclear fuel cycle such as the Thermal Oxyde Reprocessing Plant (THORP), the Sellafield Mox Plant (SMP) and Magnox power reactors.

The project for setting up a secure data transmission link between the Sellafiled site and the Commission's safeguards inspectorate headquarters in Luxembourg was been several years in the making. Persson et al. in [41] and Canadell et al. [42] present the project in considerable detail. This paper will limit itself to recall some of the aspects related to information security in those references. Already in 2006 it was possible to send 'state of health' information on the safeguards instruments installed at the facility and in the summer of 2007 the routine transmission of raw data started. Today, the procedures have been tested and routinely used by the inspectors in charge.

Early on, the SMP plant had been identified, as a test bed to implement a system of remote monitoring for safeguards, because of its modern layout. At highly automated plants such as SMP, several data acquisition systems are run in unattended, fully automatic modes. An internal network linking all detectors had been installed already at the time of plant construction. For unattended measurements, two sets of information are needed: declarations on movements of material through the process and data from in-plant safeguards instrumentation. At SMP, both sets of information are available in electronic format for automatic verifications.

At the plant, dedicated safeguards instruments and surveillance systems are controlled, operated and interrogated by the Commission via a network segregated from the operators' own systems. Data Acquisition Modules running on local PCs collect the instrument data, and copy it to a central server located in the onsite inspectors' office – where, traditionally, local workstations are used for onsite evaluations –. Acquired data includes images from the Commission's surveillance systems as well as signals from the Commission's measurement instruments – e.g. neutron coincidence counters, gamma spectrometers, identity readers –. At the same time, the Commission's systems collect data from the operator's plant control system. All this data is collected virtually in real time and – via dedicated networks – transferred to the workstations in the Commission inspectors' on-site office.

Details on nuclear material quantities, locations and movements are considered sensitive in all member States. Demonstrating the security of the system employed for the transfer and handling of safeguards data is essential.

On the plant side, all data is collected, held and processed in a dedicated Commission-controlled network, segregated from the operator's networks and systems. The measurement and surveillance data required for evaluation and assessment is then passed in encrypted form through a secure Virtual Private Network (VPN) built using internet connections. Data is in this way securely forwarded to a dedicated workstation inside an access controlled room in the Commission Luxembourg headquarters. As needed, data may then be transferred across an air gap into the specially protected local network where Commission inspectors carry out studies and evaluations.

This point-to-point secured data tunnel is built using defence-grade security encryption units. The only communications allowed are between the on-site Commission-controlled network and the Luxembourg headquarters. Integrity is further ensured by dedicated security features in the various system components which protect against external attacks. The complete system was scrutinised and accepted by the competent bodies in the United Kingdom and in the Commission.

Besides accredited secure encryption, it was also possible to agree on a coded data structure which avoids simultaneous transmission of material quantities and locations. Once the data has been securely transferred into the servers on the protected network inside Commission premises, inspectors may sort the declared balances and operations to the specific locations. This approach makes it possible to limit the security classification grading used for the information during transmission.

Using agreed security procedures, the operator provides files containing daily declaration of all pertinent moves, as well as a daily stock file. These are the same files as the operator would otherwise provide for the inspectors on site.

Thanks to the agreements reached and the systems put in place, declarations and raw data are now routinely sent to the Luxembourg headquarters. Commission inspectors are able to analyse this data and highlight issues before coming on-site. Since Commission inspectors carry out the data evaluation at headquarters, they can resolve as many issues as possible beforehand, simply by contacting the operator by phone or email. This also gives the operator more time to resolve potential issues before the inspection. When inspectors arrive at the installations later on, only those issues which can only be clarified in situ remain. The time spent on site can therefore focus on more essential discussions, meetings with the operator and other essential tasks requiring inspector's presence like design verifications and other physical verifications.

The availability of remotely transmitted safeguards data also helps reducing inspection frequencies and lower travel burden and time pressure on the inspectors. As most data can be treated and analysed in Luxembourg, it also allows better preparing the inspections.

The use of remote data transmission has made it possible for the Commission to optimise the frequency of inspections without reducing the quality of the inspections at SMP. Moreover, the Commission is working to build the next generation of standardised inspection software that will leverage even further the remote availability of data.

4.3 Exchange of information with the IAEA

The basis for exchange of information with the International Atomic Energy Agency (IAEA) is an Agreement for cooperation between the European Commission and the IAEA concluded in 1975 [37]. Article III of that Agreement provides that "The Agency and the Community shall undertake a full exchange of information and documents" and that this exchange shall be "subject to such restrictions and arrangements as may be considered necessary […] to preserve the confidential nature of certain information and documents".

The scope for the data to be exchanged between the European Commission and the IAEA is established in a number of texts:

- The Commission Regulation on the application of Euratom safeguards

– The Agreements for implementation of IAEA safeguards in the European Union
– The Protocol Additional to the Agreements for implementation of IAEA safeguards in the European Union

The Commission Regulation on the application of Euratom safeguards [39] explicitly provides in its Article 29 the possibility for the Commission to transmit to the IAEA information and data obtained in connection with the Regulation. This Article had been introduced in 1993 in the old regulation implementing Article 79 of the Euratom Treaty, Regulation (Euratom) No 3227/76, to allow provision to the IAEA of information obtained by the Commission pursuant to the Regulation that was beyond the information described in the safeguards agreements. When the new Regulation implementing Article 79 of the Euratom Treaty was discussed, starting in 1998, it was considered that the same arguments were still pertaining and the it remained necessary to keep that Article.

The Agreements for implementation of IAEA safeguards in the European Union (e.g. [38] for the non-nuclear weapon states) established in its Article 8 the obligation on the Community to provide the IAEA with the information concerning nuclear material subject to safeguards and the features of facilities relevant to safeguarding such material. This obligation is tempered by a requirement that the IAEA shall require only a minimum amount of information and data consistent with carrying out its responsibilities under the Agreement.

The Protocol Additional to the Agreements for implementation of IAEA safeguards in the European Union (e.g. [40] for the non-nuclear weapon states) provides n its article 14 that “Each State shall permit and protect free communications by the Agency for official purposes between Agency inspectors in that State and Agency Headquarters and/or Regional Offices, including attended and unattended transmission of information generated by Agency containment and/or surveillance or measurement devices. […]”, and that “[…]Communication and transmission of information as provided for in paragraph a. above shall take due account of the need to protect proprietary or commercially sensitive information or design information which the State concerned regards as being of particular sensitivity”. Article 15, which concerns the IAEA system for protection of confidential information, provides that “The Agency shall maintain a stringent regime to ensure effective protection against disclosure of commercial, technological and industrial secrets and other confidential information coming to its knowledge, including such information coming to the Agency's knowledge in the implementation of this Protocol.”

Further to the 1975 Agreement between the European Commission and the IAEA, the two organisations have adopted in 2008 a joint statement on reinforcing cooperation on nuclear energy for peace and development [43]. This creates a favourable background for strengthening the cooperation between the Commission and the IAEA in all matters relating to information security.

5 Conclusion

As Quist notes in [25], "There should be a definite, identifiable reason or rationale for classifying information or materials. If a reason is definite, then it should be expressible. If a reason cannot be expressed or can only be given in vague terms, then the information or material probably should not be classified."

The fact that information is protected by a security system – e.g. with security markings on documents and special handling procedures – is not sufficient in itself; the system must be recognised in law for the information to be properly ‘classified’.
The security regime established by the Euratom Treaty is unique, as it explicitly contemplates the possibility – and even the obligation – to prosecute security breaches according to the Member states’ criminal law.

For historical reasons, the security regime applicable to Euratom information – and particularly for safeguards information – is not uniform, The authors of the Euratom Treaty had foreseen a “gradual establishment of as uniform and comprehensive a security system as possible” within the European Union. Work still remains ahead to achieve this aim.

Nevertheless, the effectiveness of the present security regime is illustrated on a day-to-day basis in various cases: for the remote transmission of data between nuclear installations and the Commission's premises in Luxembourg; in the context of international shipments of special nuclear materials; and for the secure exchange of information between the Commission and the Vienna-based International Atomic Energy Agency.

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Use and benefits of remote data transmission

An operational feedback after one year of application in a MOX plant

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Abstract

Remote data transmission from the Sellafield MOX Plant (SMP) in UK has now been operational for more than one year. Instrument measurement data and status data are sent encrypted via a Virtual Private Network (VPN). The availability of these data at EURATOM HQ has helped the inspectorate to refocus onsite activities and reduce the inspection frequency.

The infrastructure for the data transfer has proven to work reliably and a large part of the verification activities is now carried at HQ, without the need to dispatch inspectors for data evaluations or to carry them out under the time constraints of an onsite inspection. In addition, the improved preparation of technical interventions seems to be another positive factor. However, certain arrangements are needed to ensure an efficient interface with the operator and the automatic treatment of the safeguards data is another important factor that requires further effort.

Keywords: Remote Data Transmission; MOX; NDA

1. Introduction

The enlargement of the European Union has increased the need to make nuclear inspections more efficient seeing that the number of nuclear installations and the amounts of material under EURATOM safeguards has risen considerably without a significant increase of available inspection manpower. Also the Commission's New Approaches and Integrated Safeguards (IS) for the IAEA are looking at making inspections more efficient and effective beyond traditional safeguards activities. Inspection activities on site need to focus on activities requiring human presence at the installations, all other activities should be performed at HQ if possible.

Not only can the concepts behind safeguards activities be looked at but also technological advancements. Remote data transmission (RDT) together with operator declarations at installations where unattended measurement stations are used can reduce the inspection activities that need to be carried out on site at the installations. However, details on nuclear material quantities, locations and movements are considered sensitive in all member States, and a safe transfer and handling of data is therefore of utmost importance.

Several data acquisition systems at highly automated plants such as the Sellafield Mox Plant (SMP) are run in unattended, fully automatic modes. If agreements can be reached to transmit declarations

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and raw data to HQ, inspectors are able to analyse data and highlight issues before coming onsite, also giving the operator time to resolve potential issues before the inspection. Any outstanding issues can be discussed at the time of inspection, thus utilizing the time of all parties more efficiently. Also the onsite constraints like time restrictions, operator, material and equipment availability disappear. The time of technicians can be used more efficiently if it is allowed to connect to the installed network at the installation from HQ as tests and trouble shooting can be done from HQ.

The use of RDT has made it possible for EURATOM to reduce the frequency of inspections without reducing the quality of the inspections at SMP. For a number of years inspectors were present every week and after an intermediate reduction to bi-weekly inspections, the frequency was reduced to every 4 weeks as of the beginning of 2008. RDT has also made it possible to reduce the inspection week from 4 to 3 days for regular inspections. For the Commission this is a significant saving but the same is true for the operator, since escorting of inspectors for plant entries, as required by the national legislation, is resource demanding. The reduction in frequency means however, that the inspectorate has to be vigilant and make sure that the reduced presence of inspectors does not lead to negligence versus the legal obligations of the operator's or simply safeguards unawareness among his staff. A reduction in inspection frequency could however also lead to a loss of detailed plant knowledge among inspectors.

2. Network Infrastructure Implementation at SMP for RDT

Existing data transmission systems make use of high level encryption devices, like the Datacryptor, which are up to defence standards. VPN solutions provide the best option. During 2006 the EURATOM network based in Sellafield was set up for complete remote data transmission (see figure 1).

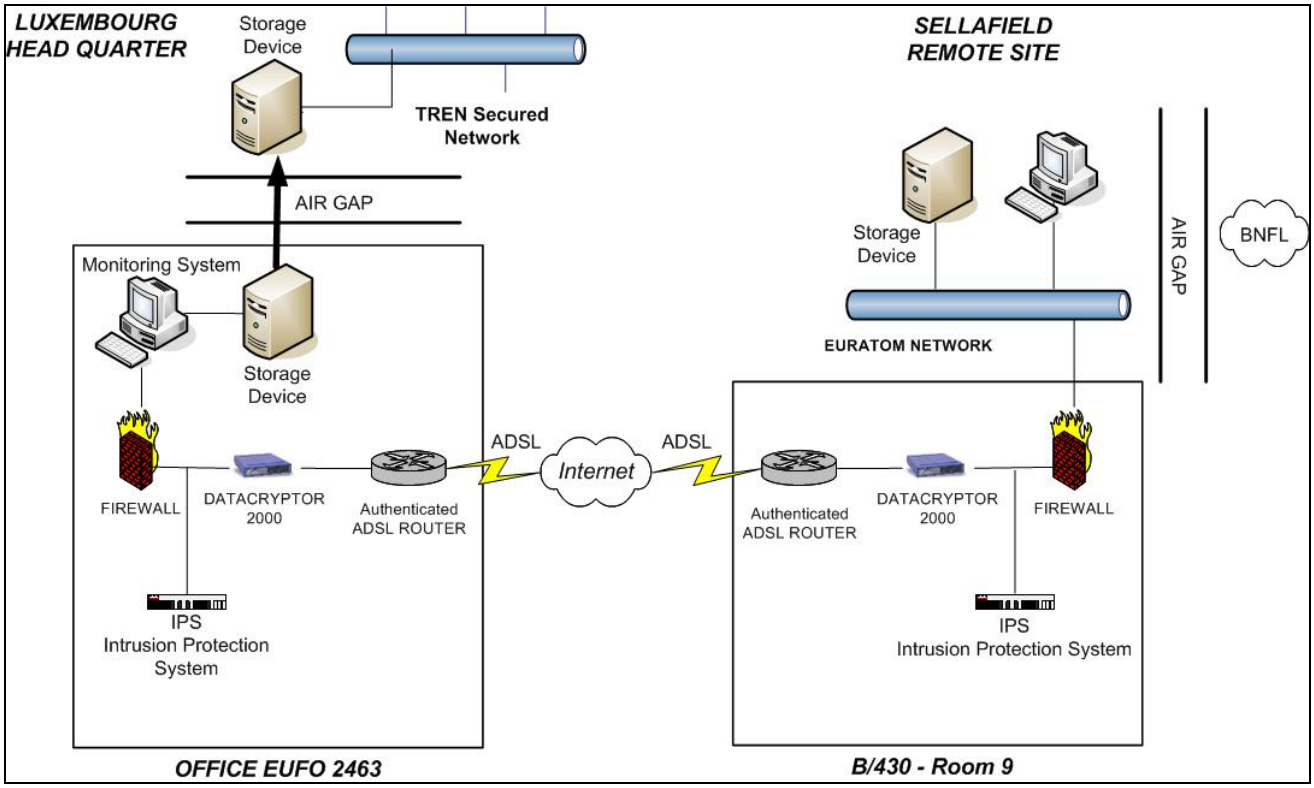

Fig. 1 Networks layout

The electronic data collected at Sellafield are held on a local Safeguards network isolated from the operating system of the main plant. All measurement and surveillance data required for evaluation and assessment purposes can be passed through the Internet to an access controlled room at the

EURATOM HQ in Luxembourg using a VPN. The secured data tunnel created with Datacryptor™ 2000 units only allows communications between the HQ in Luxembourg and the network in the on-site inspectors' office. The integrity of data is ensured by the different components of the system which prevents all external interactions. The UK and Commission services have scrutinized and accepted the system.

3. Remote monitoring at SMP

Sellafield Ltd and the EUROPEAN COMMISSION (EC) chose SMP as a test bed to implement a system of remote monitoring in the safeguards field [1] because of its modern layout. For the unattended measurements two basic sets of information are needed, the declaration on the movements of material through the process from the operator and data from the in-plant safeguards instrumentation. Both sets of information are available in electronic format for automatic verifications.

Dedicated safeguards instruments and surveillance systems are controlled, operated and interrogated via a network fully separated from the operators' communication links. Data Acquisition Modules (DAM) running on local PCs collect the instrument data, which are routinely copied to a central server located in the onsite inspectors' office and local workstations can be used for onsite evaluations. The data which are acquired include images from the Commission's FAST system, a large number of signals from Commission measurement instruments like neutron coincidence counters, gamma spectrometers, identity readers etc. but include also data branched from the plant control system of the operator, typically weight and ID readers or switches. All this information is collected virtually in real time and – via dedicated networks – transferred to the inspectors' office.

The versatility of RADAR/CRISP makes it possible to create DAMs for various kinds of signals and collect all signals using RADAR/CRISP. The combination of signals at strategic points allows for the detection of events, which are a combination of signals that are created when items pass through process locations with safeguards instrumentation installed. An event is for example a combination of time stamped signals from a neutron detector with a gamma system and a related bar code reader. It is then possible to calculate the flow of material quantities and the flow verifications consist of the subsequent comparison of the consistency checked operator data with the events detected by RADAR/CRISP. For partial defect verifications the weight and isotopic information can be extracted from the general operator declarations and loaded into CRISP.

Already in 2006 it was possible to send ‘state of health’ information on the safeguards instruments installed at the facility and in the summer of 2007 the routine transmission of raw data started. The DAMs of the RADAR/CRISP system [2] are used to record safeguards signals and technical information. The complete set of instrument data is transmitted, including raw data files, log files, alarm files and instrumentation set up files.

The log and alarm files information is transmitted in the form of simple text messages, allowing EURATOM technicians and inspectors to check whether the instrumentation is running or to establish where possible error sources or technical problems exist. The log and alarm files are transmitted once a day to HQ in Luxembourg together with the raw data files collected from installed data acquisition systems. The data are transmitted on the EURATOM network which is 'air gapped' at both ends, i.e. at the Sellafield site it is a separate network and in Luxembourg it is air gapped from the EC internal networks, resulting in a minimised security risk for both the operator and the inspectorate.

To make full use of the RDT it is necessary to obtain the operator declarations for the movements of material through the process at HQ. A simple mail box system was agreed with Sellafield Ltd in 2007/2008 for the transmission of these declaration files. It was possible to agree on a coded data structure without any direct link to quantities and locations. At HQ it is possible to sort the moves and stocks to specific locations. The lack of direct information made it possible to transmit the information without the need to classify them. Using the agreed structure the operator provides daily declaration files of all pertinent moves and a daily stock file. These are the same files as the operator would otherwise provide for the inspectors on site, the only difference being that they are sent through the email system to the EURATOM HQ in Luxembourg and no additional preparation for transmission is required.

The operator's declarations on material moves in the process are handled by software applications which check them for internal consistency and allow querying on the data provided. These applications can be run at HQ and the data checked in order to discover potential issues.

4. RDT and Inspection Activities at SMP

Inspection activities at SMP include accountancy verifications, surveillance review, analyses of unattended measurement equipment, plant visits for physical verifications, seal verifications, servicing of instruments and BTC verification. The additional physical verifications besides the verifications of the unattended measurement stations are carried out at strategic flow locations in the plant and random checks of items in process areas. To maintain continuity of knowledge of some stores, seals and/or surveillance cameras have been installed resulting in the necessity to perform seal verifications/replacements and a surveillance review with means of a FAST system. As SMP is a relatively new plant an internal network linking all detectors was installed already at the time of construction.

The data transmitted from Sellafield are normally treated 2-3 times per week in Luxembourg. Although the raw data is transferred to HQ every day the data are at present transferred to the secure network twice a week with operator declarations being received via email every day. SMP was designed to be a plant without too many manual interventions and that also goes for the operator's MPIC (MOX Plant Information Computer) system in which all normal events are generated and reported automatically. If material moves in a way not originally foreseen it is necessary for the operator to manually generate the move in MPIC. In such a case the declarations can take a bit longer and it is not always possible to analyse all events from the previous day due to a lack of declarations. Once the plant reaches full capacity it will probably be necessary for inspectors to treat the data every day unless the data treatment and analyses can be made more automatic. Analysis of data is currently carried out as needed depending on the number of moves through detectors requiring analyses.

The use of RDT has meant that the data from the many Unattended Measurement Station (UMS) systems can be analysed at HQ and time spent on inspections can be dedicated to the rest of the tasks and to have more time for discussions and issue follow ups with the operator. Minor issues can be resolved before the inspection by contacting the operator by phone or email, making it possible for the inspectors to arrive at the installations with a list of issues needing further clarifications. The time spent at inspections can thus be used in the best possible way with physical verifications, meetings and discussions and yet save resources.

The possibility to have the raw data in Luxembourg makes it possible to analyse the data at HQ and make sure that it all agrees with the operator declarations. The analyses also make it possible to spot problems in case they would not be reported in log and/or alarm files. Recently the inspectors encountered a problem with the data analysis of data from a neutron detector and by analysing the raw data at HQ it became clear that the problem was caused by one part of the neutron detector not functioning correctly. The particular detector consists of two halves and using the data readily available at HQ it was possible to quickly respond and produce a new calibration curve using the functioning half of the detector. This could be done immediately at HQ after discussing the issue with specialists who also could assist in producing a new calibration curve. Using the new calibration curve it was then possible to continue analysing data before and during the inspection.

The operator has also granted permission to send images from the surveillance system to Luxembourg. In order to obtain this agreement the operator requested to see the field of view of each camera prior to giving their agreement to make sure that what was shown on the image was not to be considered as sensitive information. The possibility to transfer images has proven useful when the operator wants to perform tests of various kinds and there is no inspector on site. In such a case it is possible to agree on a time when an inspector can look at live images in Luxembourg and simultaneously communicate with the operator and provide immediate feedback. So far, the option of transmitting images has been used only partially due to bandwidth restrictions of the ADSL lines in the remote plant of Sellafield. In the future the possibility to perform a surveillance review at HQ could further reduce the inspection work that needs to be carried out at the installation.

At present EURATOM is working on a standardised inspection software application called VARO. The aim is to check and validate operators' data with a standardised application and to have an automatic information exchange with the measurement applications, mostly RADAR/CRISP.

Figure 2 shows the intended future implementation of software and hardware tools in SMP integrating the use of automatic verification tools and remote data transmission.

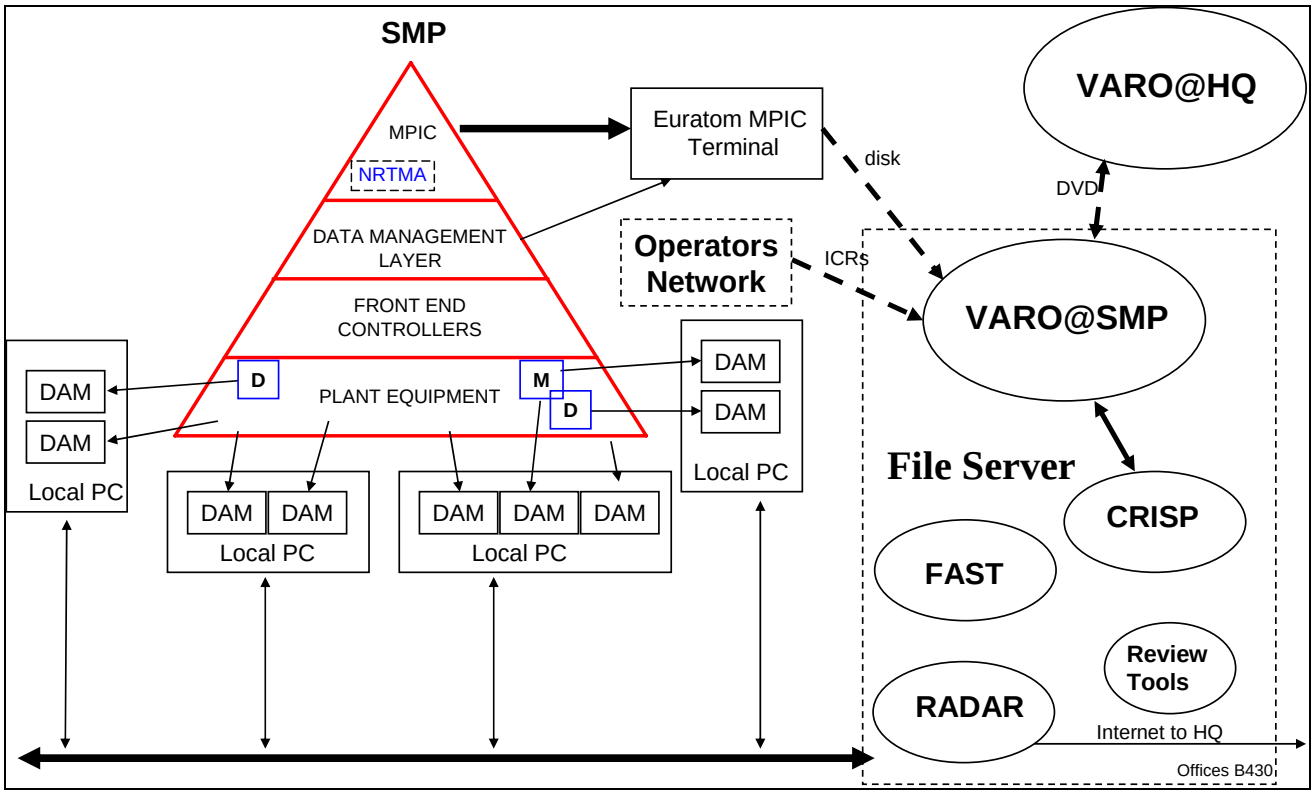

Fig. 2 Implementation at Sellafield EURATOM offices

If all information described above is available at HQ, most inspection activities could be performed independently on-site and in HQ. The need for further random verifications in the process areas, discussions for issue follow-up, meetings and the overall performance of the operator and safeguards awareness among the personnel could decide the number of actual visits needed.

5. Advantages of RDT

The availability of remotely transmitted safeguards data has not only significantly reduced the inspection frequency but also the travel burden and time pressure on the inspectors as most data can be treated and analysed in Luxembourg. It has also allowed for a better inspection preparation. The inspectors carry out the data evaluation at HQ in preparation for planned inspections, resolve as many issues as possible beforehand by contacting the operator by phone or email and arrive at the installations with a list of issues needing further clarifications. The time spent on site can focus more on essential discussions, meetings with the operator and other essential tasks requiring inspector's presence like design verifications and other physical verifications.

The availability of instrument data at HQ has also allowed for a better preparation of planned interventions or breakdown maintenance. The inspectors and/or technicians are able to prepare on site activities by checking the instrument data, state-of-health messages available and by analysing data to confirm operator declarations. In case of instrumentation problems specialists are readily available at HQ for trouble shooting and advice, making it sometimes possible for the inspectors to solve the problem at the next visit without the need to send specialists out for minor works. Using the remote control of instrument parameters is also a very useful tool to minimise on site interventions as

parameters can be modified and software upgrades done via the network link. In case there is a need for instrumentation to be shut down for short periods after a request from the operator, it is possible to do so from Luxembourg instead of the operator having to wait for an inspector to be present.

The availability of data from the onsite systems at HQ allows the use of common review tools. Applications like RADAR/CRISP [2] already have the functionalities to deal with data from different installations and other tools like VARO are in development.

Centralising evaluation routines make it easier to use and maintain software applications without having to send technicians for maintenance or to use inspection resources for these interventions. All software upgrades can be first tested at HQ and then implemented on site.

One of the major challenges in safeguards is the training of inspectors on the variety of tools used. The availability of site data at HQ allows for the training of inspectors with real data instead of faked set ups that often only give a blurred picture of the situation at the installations. Moreover, with the availability of inspection software tools at HQ, the use of standardised evaluation routines and criteria will be more easily implemented and generic software applications developed consequently. This harmonised approach will allow for more efficient inspection activities and improved training of inspectors.

Finally, one of the common goals of any inspection based organisation is the optimisation of the use of manpower. Sending an inspector to an installation is causing a considerable overhead both in terms of travel costs and time. All evaluations that can be done at HQ have in addition to the saved travel overhead the advantage that the inspector is not under time pressure to finish the verifications in the timeframe of the plant visit. This leads to more flexible arrangements, decouples the depth of safeguards evaluations from the time spent at an installation, and helps to achieve better overall inspection results.

6. Conclusions

Successful and comprehensive safeguards have always been built on co-operation of all parties involved. Remote Data Transmission of sensitive safeguards information and signals is another example of how result oriented co-operation can help to make safeguards inspections more efficient and effective without compromising the quality of the evaluations. At SMP the implementation of RDT has meant that EURATOM has been able to reduce the frequency of inspections without compromising the quality, efficiency and effectiveness of their verifications. In addition also the operator can use his personnel more efficiently.

The described approach is just an example of a possible implementation scheme and the Commission remains open for discussions on alternative implementation schemes as long as the main goal to detach data evaluation from site presence is achieved.

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Information Surety for Safeguards and Nonproliferation

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Abstract:

A systems-level approach is necessary for managing both safeguards- and nonproliferation-relevant information. Such an approach identifies why, where, when and how to incorporate tamper indication, authentication and encryption in both safeguards and nonproliferation information systems. The need encompasses the generation, communication, retention and storage, potential further distribution and sharing of information. Requirements are driven not only by the needs of intended audiences, but also by threats of disruption, interception, or manipulation by unintended parties. Many issues are involved, including, but not limited to, timeliness, completeness, provision by "pushing" or "pulling" data, problem resolution, information barriers, proprietary concerns, and multilateral sharing. Successful implementation of system-level information security is an essential, fundamental aspect of establishing and maintaining trust between cooperating entities generally, and in providing nonproliferation assurances specifically.

Keywords: information, nonproliferation, safeguards, security, surety

1. Introduction

Information about nuclear materials, facilities, operations, research and development, including information used for international safeguards, is increasingly seeing broader dissemination for various purposes, such as “transparency.” What effect, if any, does this situation pose for international safeguards—and for interests generally to assure the nonproliferation of nuclear weapons?

We examine the high-level considerations involved in providing information for the sake of strengthening the nonproliferation regime, whether for IAEA safeguards specifically or for nonproliferation ends generally. We introduce the concept of “information surety” to describe the relative value of information to create or build trust.

2. Need for a Systems Approach

2.1. Role of Information for Safeguards

International safeguards constitute an information system: The system involves the collection, storage, retrieval, transmission, analysis, and evaluation of information. That information comprises many varied sources, including declarations, measurements, inspector observations, published documents and other “open source” materials, reports from remote monitoring, and much more. The International Atomic Energy Agency (IAEA) analyzes the information to come to an assessment about material non-diversion and the completeness of a state’s declaration.

Thus international safeguards constitute a decision support system, specifically. The “decision” for international safeguards is effectively the *conclusion* that the IAEA reports annually to its General Conference of member states for each state with a Safeguards Agreement.

Ideally, the safeguards decision support system would suffice to assure the world that each state is a responsible user of nuclear energy for peaceful purposes. Ultimately, the system depends on the trust that the member states accord to the IAEA. However, the system does have possible shortcomings. For example, the IAEA only shares its safeguards conclusion with member states. Many of the supporting details are not shared, because they must be protected as safeguards confidential. To protect the IAEA Secretariat from influence and manipulation, its internal workings are largely independent and not accessible directly by the member states. Even the safeguards conclusion is updated just once a year, which in some situations might not satisfy a member state.

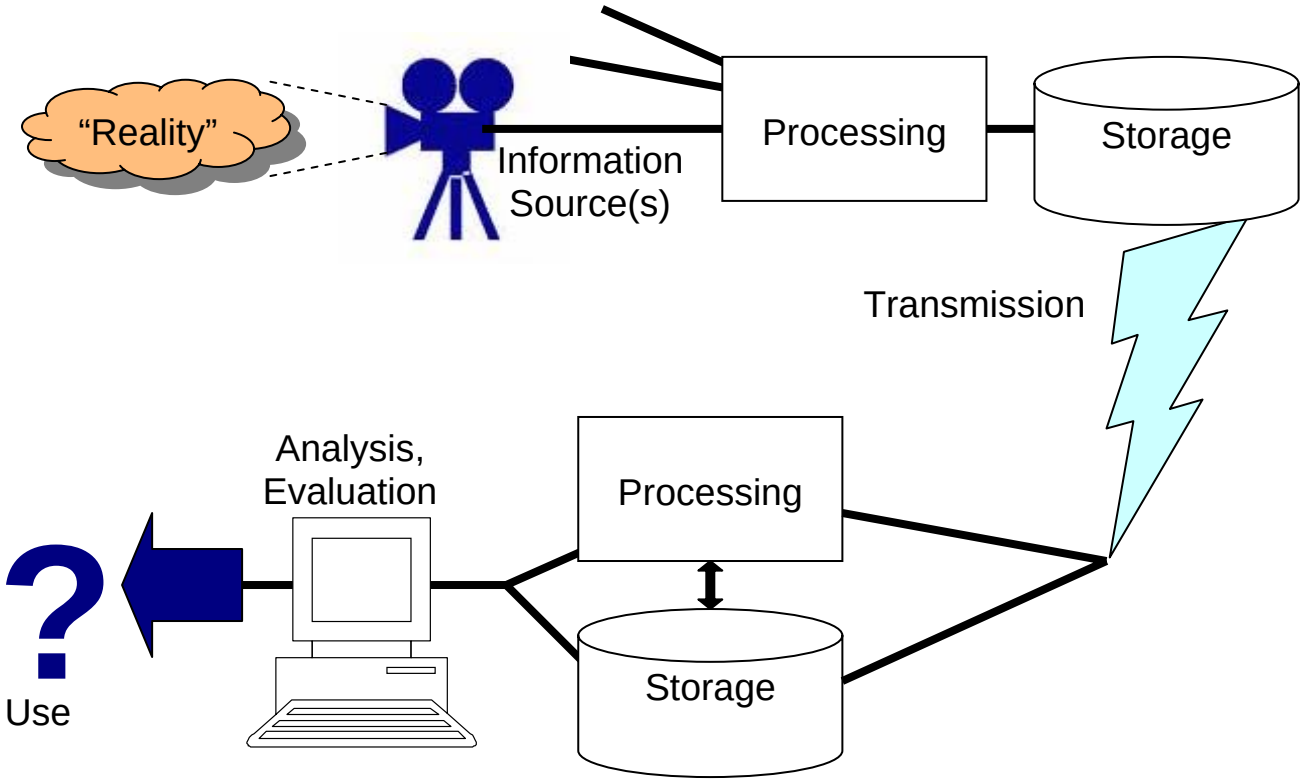
2.2. Role of Information for Nonproliferation

Given the critical importance of the potential threat, it is not surprising that other approaches have arisen in the interest of strengthening the nonproliferation regime. Regional authorities may complement the IAEA's safeguards system; indeed, Euratom has been in existence just as long as the IAEA. There is a continuing struggle to balance the need for a robust nonproliferation regime, while not unnecessarily duplicating the regulatory burden and thereby driving up costs.

Another trend has been the emergence of "transparency measures" by states or facility owners to release a variety of information directly and voluntarily to outside audiences. Sometimes neither the end use nor the audience is clearly defined. What is worse, it may not constitute a decision support system at all, but a relatively vague "information provision system." It is therefore important to view even these non-safeguards or quasi-safeguards measures using a systems-level approach.

Such transparency measures are usually well-intentioned, and offer the prospect of adding the desired "robustness" to the nonproliferation regime. Nevertheless, there are risks. Implementation issues are critical to ensure that the provision of information indeed contributes to the goal of addressing concerns and building trust. Freely-dispensed information by itself does not guarantee that end. In the worst case, it can even be counterproductive and damaging.

2.3. The information system



Although the illustration here is oversimplified, it is a reminder that there is path between an information source and its destination with many opportunities for its content to change or be changed. It also suggests potential issues at the endpoints: There is a gap between the reality itself and the source of the information; likewise, the end use of the information is not always clear. Yet another dimension, only suggested here by the "storage" elements, is that of time dependence: delay in forwarding information to its end user, and the retention of historical information.

3. Information Surety

Surety is defined by Merriam-Webster as "1: the state of being sure: as a: sure knowledge : CERTAINTY". For the purposes of this discussion, we extend the concept to define

> Information Surety: a measure of the degree to which information can be trusted.

It is important to keep in mind that information, *per se,* is merely a representation of or a report about reality; it is not reality itself. It is not inherently the case that information reflects the truth, nor is it a simple matter of information being true or false. Unfortunately, there is no clear *quantitative* measure of information surety.

3.1. Elements of Information Surety

Several factors determine whether information is worthy of trust. From the point of view of the person receiving the information, the following characteristics are important:

- Correctness, accuracy ("the truth")

  If the information says it's a ton, then the object truly weighs a ton. Although seemingly obvious, there are instances where the fact isn't as easily established. The calibration of a measuring instrument is involved in establishing accuracy.

- Completeness ("the whole truth")

  Partial truths can actually lead to misinformation. Although the fact that a certain cask ID had been shipped may be correct, information surety would require us to know also that it happened to be one of *four* casks that shipped, for example.

  Completeness can require special approaches in practical situations, however. Sometimes a brute force collection of information on a total population of items of interest would be unrealistic, such as conducting quantitative radiation measurements on every individual fuel assembly in a spent fuel storage facility. In such situations, a sampling approach will often suffice for completeness, provided that the sampling is done randomly and for an adequate sample size.

- Relevance ("nothing but the truth")

  Provision of irrelevant information may lead to confusion, or obscure meaningful evidence in a sea of distracting noise. Certainly there is no added value from the provision of irrelevant information; at worst, it may be indication of deceit.

- Time

  When is, or was, the information true? Many situations are dynamic, so information is usually "correct" only at (or for) a specified time. Time is therefore often an essential piece of information surety. In some sense, it may be considered part of the completeness requirement.

  Time also may involve *timeliness*, in that the value of the information may change (usually decrease) with time. However, timeliness generally should not affect information surety: i.e., whether or not the information can be trusted.

  In some cases, information serves a dual role: it may be used both as the basis of a state's safeguards declaration as well as by the IAEA to verify that declaration. Information timing is critical to whether or not such dual use is acceptable. For example, the operator should not know the measurement data that will be used to verify a declaration, before the declaration is made.

- Uncertainty

  The estimated precision of measurements, the confidence in subjective assessments by inspectors, and other measures of uncertainty in the information, are essential characteristics from an information surety perspective.

- Source

  How the information was created is important to surety: Was it measurement equipment? Owned by the information recipient, by the operator, or perhaps by a third party? What ensures that its output is believable? How is it protected? How are we confident that even the source of the information is able to convey the reality of the situation? For example, is a video camera indeed showing us a real scene, or is there just a picture in front of the camera lens?

  In other cases, information might be more subjective, such as the written observations of an inspector. Then the question becomes one of *who* made the observations? What qualifications, experience and training did they have?

  Just as we care about the quality or trustworthiness of references in a technical report, information supplied for safeguards and nonproliferation must originate from credible sources.

- State of Health

  Particularly in the case of electronic instrumentation, one worries whether the source is able to provide the intended information. Does the instrument have power? In the case of a camera, is the scene illuminated? Some sources would normally only send information if they detect a problem: a fiber-optic seal opening, for example. For this reason, we require "state of health" messages from the seal (that cannot be impersonated) so that the system can confirm that the seal is functioning. Information surety requires that we can truly be confident that "no news is good news."

- Authenticity

  Authenticity ensures that information hasn't been substantively modified between the source and the recipient, whether the change was unintentional or intentional. Authenticity ensures us of the *integrity* of the information. It further confirms the source of the information; that the source has not been "impersonated" by a fraudulent one.

  Cryptographic measures are often required to validate information authenticity. Ideally they are applied as close to the information source as possible, by combining the content of the information with a "key" to produce an authentication tag via a standard algorithm. The tag then accompanies the information wherever it goes. A recipient with another key is able to verify the authentication: i.e., determine whether the received information is consistent with the authentication tag. Unless authentication measures are employed effectively, information is vulnerable to many different kinds of attacks or simply to accidental corruption.

  Authentication measures can further help to ensure that multiple stored copies of the same information are truly identical.

- Abstraction

  Information is rarely passed on without some degree of initial processing, classification, or compilation. By "abstraction" we mean any transformation of raw data into the information that actually gets transmitted, stored, analyzed or evaluated. ("Data reduction" is sometimes another term used to describe this step.) Some abstraction may take place close to the sensor; more might occur close to the information recipient. Abstraction is both valuable and necessary to deal with otherwise unmanageable loads of information. Nevertheless, information surety demands that such transformations do not introduce substantive changes.

3.2. "Extended" information surety

The information surety elements we describe above really only consider the point of view of the *receiver* of the information. However, a more balanced viewpoint would include other considerations, especially elements that may be important to the *provider* of the information. We therefore use the concept of "extended" information surety to refer to such two-way surety assurances.

Especially for sensitive information, the provider's willingness to share it often depends on how it might be handled and used after disclosure. In many cases it should not be available to anyone, so the intended audience needs to be defined. To prevent the disclosure of information disclosure to others, extended information surety incorporates encryption. The management of "keys" that enable one to decrypt the protected information, and the control of those who have privileged access, are important elements of extended information surety.

3.3. Validation of Information Surety

Validation would answer the question, "How do I establish the truth of the information I receive?" How can I be confident that the information is correct, complete, relevant, and authentic? Various means may be involved, and not all provide the same degree of confidence:

- Access

  Perhaps the most convincing information is what we are able to witness directly, without any intervening technology or people. When one is able to use eyes, ears, and other senses, possibly supplemented with various trusted equipment in one's possession, the information system becomes a much simpler problem. For this reason, onsite inspection is unlikely to disappear: It offers a relatively high degree of information surety.

- Verification

  Occasional tests can help to ensure that the routine provision of information is

trustworthy. Automated systems for remote monitoring, operator declarations and the like can be more believable if they are subject to direct inspection that is unannounced or random. Even if an inspection doesn't actually happen, just the possibility that it could happen provides some increased confidence for information surety.

- Secure systems

  The use of protected, trusted equipment that remains under the control of the information recipient or that of an independent party is more likely to supply trustworthy information than equipment controlled by or accessible to the facility owner. Security measures can protect the information channels.

- Redundancy

  Clearly information surety is improved whenever multiple, independent methods are employed to acquire the same information. The information system of course needs to check that the resulting information is the same for all methods.

- Consistency with expectations

  There may be an a priori basis for what information to expect. Shipper/receiver differences are an example from safeguards: information about material leaving one facility should cross-check with material entering another. Information reported periodically often is not expected to vary significantly from one report to another. However, preconceived notions of what the information "should be" actually can make it easier to accept a lie.

- Inherent characteristics

  The perceived difficulty of falsifying information is another basis for information surety. For example, as a given object or process is viewed with increasingly many different information channels, especially when they are related in complex ways, our information surety about the larger picture improves. It becomes less likely that any individual input makes a critical difference, and any attempt to deceive would require coordinated attacks on multiple inputs.

- Historical experience

  If in the past there have never been any concerns about the trustworthiness of information from a given provider, then there is greater confidence in believing other information from the same provider. The consequences of a breach in the record can be substantial. Once that has happened, everything from the same provider is suspect, even if for no other reason.

- Cooperation

  Even without any past experience, a provider who is more forthcoming and willing to supply information is often regarded as more trustworthy than one who obstructs. This aspect is not necessarily a dependable indicator of information surety, however.

4. Related Issues

4.1. Other Requirements

Other issues may affect the information system used for safeguards and nonproliferation, yet do not affect information surety. For example, the information system and the information it produces can do no harm: safety and security considerations by the provider cannot be ignored. There may be other constraints, such as noninterference with operations, and minimal intrusiveness.

There is invariably a tradeoff involved, between the minimum information required to obtain a conclusion (information is sufficient), and attempting to maximize the amount of information provided (perhaps an objective for transparency). The minimalist approach would be ideal, but it likely implies reduced information surety.

4.2. Threats to information

Information is at risk from both accidental causes and intentional ones. System failures can happen that lead to the loss or corruption of information. They might also happen without any loss in information integrity—for example, the failure might be to an encryption process, resulting in the unintended exposure (compromise) of the information.

Eavesdropping presents a threat to information deemed confidential to the provider and intended recipient. In this situation, the information is not changed or obstructed;

instead, it is able to be seen by unintended outsiders. Encryption, tamper indication, and keeping other parties physically removed from the information system are ways to mitigate such threats.

Intentional "tamper," when a malicious outsider deliberately alters information or the information system itself, is a threat that goes one step further than eavesdropping. The motivations can range from the desire to provide misinformation, to obstruct information delivery, or simply to vandalize. Tampering may leave obvious indications, subtle ones, or possibly none at all. Tamper indication and data authentication methods are designed to ensure that tampering cannot go undetected.

4.3. Use of information

Extended information surety wants to know more about what happens to the information once it's "out." Who acts on it? What will be done with it? If some evaluation was done, what was the reaction or response? At least, what did they think about it? If there were any sensitivities, and the information had been disclosed in confidence, will those sensitivities be respected? Who else might have seen the information?

Where information is not required for compliance, but offered for other reasons, there are additional questions: What is wanted or needed? Why would one provide the information: does anybody care? Exactly who is the audience? Does the recipient just want to "have" it, rather than actually evaluate or do something with it? For transparency applications, is it the mere act of providing the information that's important, or is it the information content itself? What are the risks involved? How might the information be misinterpreted or misused? Might it be shared beyond its intended audience?

Information systems that support multilateral, rather than simple two-party sharing, can present significant complications. In multilateral arrangements, the threat can exist from within the sphere of collaboration.

For safeguards and nonproliferation applications in general, it is essential that one considers the possibility that information may not always turn out to be as expected. What happens when the message turns out to be bad news, or possibly a mistake? There must be mechanisms in place to resolve anomalies, ambiguities, and problems, before they lead to a breakdown in trust. Such mechanisms are the "circuit breakers" that provide strength and resilience to the information system.

5. Conclusion

The issues are too extensive and the situations too varied for us to address information surety comprehensively. What is clear, however, is that information is not inherently trustworthy. It is necessary to analyze, examine and evaluate a comprehensive picture of an information system from its source through its final disposition.

Safeguards have evolved, and continue to evolve, an extensive system for processing information. There are pressures on the system to reduce intrusiveness, to minimize costs, and even to enlarge the scope—it is important that changes to the safeguards system are done mindful of their impacts on information surety. Particularly with the transition to "information-driven safeguards," it is essential to consider the *surety* of safeguards information, not merely the information itself.

Other information systems for nonproliferation ends should similarly address the complex implications of information surety.

6. Acknowledgment

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SESSION 17

REPOSITORY AND WASTE

Vulnerability assessment of safeguarding a geological repository

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Abstract

In the last 15 years several IAEA working groups have developed safeguards approaches for the various stages and parts of geological repositories. In this paper we will limit our evaluation to the last stage of the life cycle of a geological repository, i.e. the operation and the final closure of the repository. We summarise the diversion scenarios for a geological repository and we propose a methodology to assess its vulnerability, depending on the safeguards measures undertaken. Vulnerability can be assessed for each scenario based on three criteria: detection probability, false alarm probability and delay time. Details and practical implications are discussed based on an illustrative example.

The assessment will result in recommendations for an effective and efficient safeguards approach for geological repositories.

Keywords: vulnerability assessment, geological repository, safeguards

1. Introduction

There are several concepts for a geological repository and the conditioning plant that is used for preparing the spent nuclear fuel before it is stored underground. The concepts differ with respect to disposal medium, choices regarding the geometry of the tunnels and underground and surface facilities. Differences in the conditioning plant design are related to e.g. their location with respect to the repository and the type of container that is used for final disposal.
In the framework of the IAEA task SAGOR generic concepts of the conditioning plant and the geological repository have been developed, where the latter is called the reference repository. These concepts are composites of the different existing designs.

Based on the generic or reference concepts, diversion paths for nuclear material (spent fuel) have been developed in order to analyse the proliferation resistance of the facilities and to develop a safeguards regime that covers all diversion paths.
In this paper we propose an analysis of the proliferation resistance based on vulnerability assessment. In the safeguards context, vulnerability will be understood as lack of capability to (timely) detect a diversion of fissile material, the objective of the vulnerability analysis being to determine which safeguards measures are able to ensure a sufficient degree of resistance to diversions. This requires first a characterisation of the (type of) geological repository considered, the diversion scenarios and the possible (combinations of) safeguards measures. Subsequently, a multi-criteria approach may be employed to evaluate the vulnerability of a specified geological repository (open or closed) in the context of various safeguards measures, for each diversion scenario considered.

In the next section we give a general description of an open (or operational) and a closed geological repository, respectively, while in section 3 we summarise the relevant diversion scenarios. In section 4 the methodology proposed for evaluating the vulnerability is introduced and subsequently detailed on an illustrative example. Section 5 contains a discussion of the obtained results and practical implications. Ideas for further extension of the methodology are sketched out in section 6.

2. Generic description of a geological repository

2.1. Description of the open geological repository

The description of the reference geological repository was developed during SAGOR's phase 1 and is based on Wuschke [1]. The description is divided in two parts: a part describing the physical properties of the site and a part describing the activities at the site. In this paper a summary will be given. For the full description the reader may consult the above-mentioned reference.

2.1.1. Physical properties of an open geological repository

A geological repository consists of a large number of disposal rooms, access tunnels to these rooms, shafts or ramps for access from the surface, ventilation shafts and supporting surface and underground facilities. It is designed for disposal of spent fuel at a depth of about 500 m in crystalline rock. Other geological materials that are considered are salt, tuff and clay.

The surface area of a repository is divided in a supervised area (nonradioactive) and two potentially radioactive, protected areas. The main protected area contains a cask reception/transfer area, a service-shaft complex and some ancillary services. The conditioning plant is located on-site in the reference design. The second protected area contains the upcast ventilation shafts, headframes and possibly other services.

Surface facilities include the cask reception/transfer area where casks are received that come from the conditioning facility and are transferred to the waste shaft headframe. Other functions of the cask reception/transfer area are the storage of a sufficient number of full and empty casks to ensure continuous operation and the backtransfer of empty casks to the conditioning facility.
The waste shaft headframe houses the top of the waste shaft and the cage with which the casks are transferred underground.
Other surface facilities include facilities for the receipt, transportation, storage and preparation of materials for the preparation of buffer, backfill and concrete and waste management facilities.

There are various types of shafts with different functions, like the waste shaft, service shaft and ventilation shafts.

The emplacement level is designed as follows. Rooms for the emplacement of spent fuel are arranged in panels that are connected by twin access tunnels and twin panel tunnels and surrounded by a perimeter tunnel. A panel is a group of disposal rooms excavated from a common access tunnel (see figure 1). At the emplacement level a hot cell facility is located for handling e.g. damaged canisters and a cask storage area with a capacity of a few-days throughput of casks.

Exploration and monitoring boreholes have been drilled during site selection and characterisation that intersect the emplacement level. Some of them might end outside the protected areas. However, the diameter size is usually not larger than a few centimeters. Those with larger diameter are very likely to be in the neighbourhood of the operating repository shafts and therefore within the protected area.

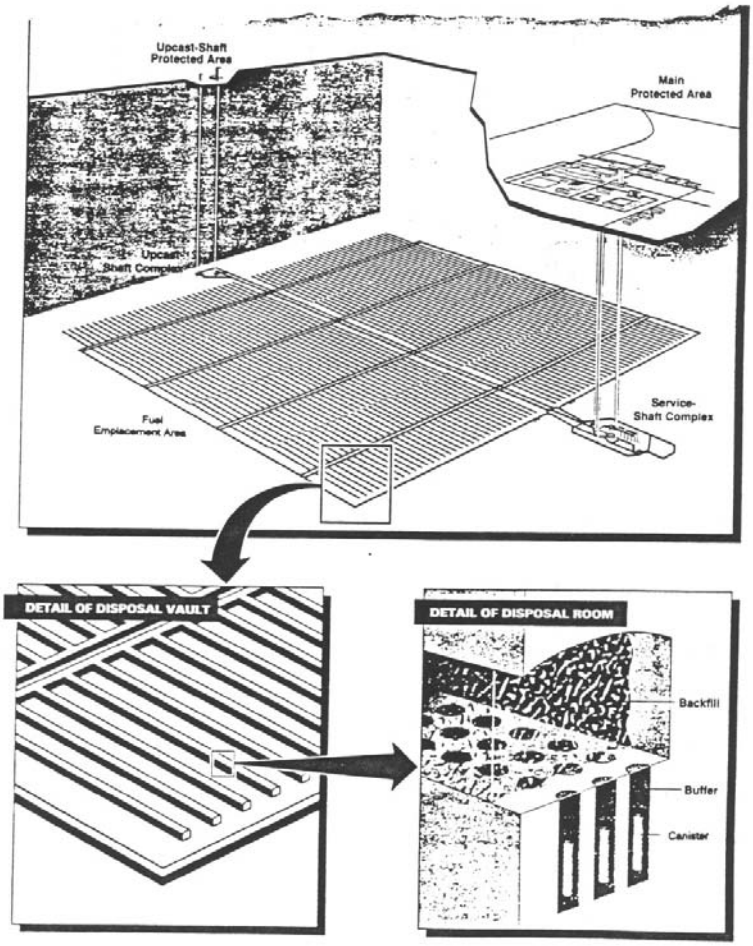


FIGURE 1: The Reference Repository

2.1.2. Activities in the open repository

The basic items that are handled in the repository are the disposal canister and the transport cask. Although the repository design and operation is mostly independent of the spent fuel characteristics, the size and mass of the spent fuel assemblies affect the size of the canisters, while the heat output influences the spacing of canisters in the repository.
The disposal canister has an outer canister of copper for protection against corrosion and consists internally of steel for strength. It is designed for containing 9 BWR fuel elements or 4 PWR fuel elements with a mass of approximately 2 tonnes. The canister is 5 m high and 1 m in diameter and has a mass of 18 tonnes without the spent fuel. The lid is welded and a unique identification is foreseen. Other canister designs consider titanium or only steel.
The transport cask is designed to reduce the radiation levels for the workers to acceptable levels (a few µGy/h in contact). It has a mass of 35 tonnes while empty. Provisions are made for the application of safeguards seals.

Operations that take place in the repository are e.g. excavation of rooms, installation of equipment, receipt and emplacement of canisters in boreholes, backfilling of boreholes and storage rooms.

At the end of the activities the repository will be decommissioned and closed.

2.2. Generic description of a closed geological repository

The decommissioning of the repository includes backfilling and sealing of all boreholes, tunnels and shafts. At strategic points concrete bulkheads will be placed. All underground facilities and equipment will be either decontaminated and removed or abandoned. Subsequently surface facilities will be decontaminated, dismantled and removed. Monitoring equipment will remain as the authorities deem it necessary.
The surface will be landscaped or reforested and made suitable for public access. Permanent markers are installed to signify the presence of the geological repository and its content.

3. Diversion paths of fissile material

3.1. Diversion paths in an open geological repository

Based on the final SAGOR report [2] some diversion paths have been considered with the associated anomalies. These are discussed in this section. This consideration is performed following the sequence of the handlings of the spent fuel. Anomalies with respect to operator's records and reports have not been taken into account.

Open repository

1. During transport to the reception area transport casks with spent fuel can be diverted. The most sophisticated way to conceal this is to replace a transport cask by a dummy containing radioactive material and to tamper the seals. An alternative is to circumfere the seal by opening the cask and rewelding it away from the seal. Anomalies to be observed can be tampering with seals, inconsistencies in measured radiation of contents cask, welds at unexpected positions at the cask.
2. During presence in the buffer storage similar diversions can be envisaged as mentioned for the period during transport. This type of diversion will be more difficult to perform due to the limited space, the possible presence of an inspector and one can also imagine that a camera has been installed to monitor the movements of the casks.
3. During transport from the reception area to the underground disposal room a diversion can take place in the same way as in 1. A complicating factor for the diverter is the point after the container has entered the shaft: the container then has a natural barrier, with only the shafts as entrance/exit. Anomalies are the same as in 1, plus an undeclared transport of a (partially) filled container (e.g. with fuel rods) out of the shaft or reprocessing activities in the underground area.
4. After placement in the disposal room the container can be recovered from the borehole and removed. Several options exist for the diverter:
   - the diverted material is reprocessed underground and only Pu is transported to the surface via e.g. shielded overpacks
   - the container is transported to the surface via an existing shaft
   - the container is transported to the surface via an undeclared shaft
   - the content of the container is transported in small amounts to surface via "empty" casks

   Anomalies are undeclared mining activities, undeclared underground reprocessing facility, undeclared transport of (partially) filled container, cask or other equipment to surface or an undeclared extra shaft (may be concealed by normal excavation operations), inconsistencies in measured radiation of contents cask.

3.2. Diversion paths in a closed geological repository

After closure of the repository a diversion can only take place by excavating containers and bringing them to the surface. Therefore mining activities are required, either by reopening the original shaft, by creating a new shaft or by approaching the containers underground. Anomalies are undeclared mining activities. The following diversion paths have been defined:

1. Reopening the original shaft. Reopening the original shaft may be interesting for harder geological media like rock and tuff with respect to the time needed for the operation. For the more plastic media it will probably be faster to create a new shaft.

2. Creating a new shaft. Creating a new shaft can be an alternative to reopening the original one, especially for the more plastic media like clay or salt.

3. Approaching the containers from an adjacent mine. This diversion path will take more time, but has the advantage for the diverter that there will be no visible anomalies on the earth surface.

Instead of shafts ramps can be constructed, too.

In the framework of SAGOR an analysis of the time required for several scenarios was performed in order to link this to the timeliness criteria for inspections. In the course of the last ten years excavation technology has improved, resulting in higher excavation speeds. It has been therefore concluded that such an analysis should be performed at the start of the operational phase of the repository and repeated at regular intervals.

4. An approach to vulnerability analysis of a geological repository

4.1 Background

Vulnerability is recognized in the literature as a difficult multidimensional concept, for which the indicators are most often described only in qualitative terms. In the latest years however a number of quantitative vulnerability analysis approaches have been proposed, for instance by Gheorghe and Vamanu [3] and Ezell [4] for complex systems and/or critical infrastructures. In the latter study it is argued that vulnerability assessment is a component of risk assessment. While risk assessment considers the triplet (scenario, likelihood, consequences) and focuses on likelihood and consequences, vulnerability focuses on the *susceptibility to a scenario*.

For safeguards aims, the likelihood of a scenario is not of particular relevance since all diversion scenarios should be covered equally well by the safeguards approach, regardless of e.g. budgetary implications. A diverter will probably choose for the scenario with the lowest detection probability if this would exist.

In order to cover all diversion scenarios equally well with a safeguards approach, vulnerability is a useful concept to evaluate different safeguards approaches with respect to their power to detect diversions done via the different scenarios. The focus of vulnerability on the susceptibility to a scenario makes it more safeguards-relevant than risk assessment with its focus on likelihood.

To address the multi-dimensionality of vulnerability and the interaction between the different influencing factors, multi-criteria decision aid tools have been employed in a number of studies. Accordingly, the global vulnerability V of a system is often defined through a multi-attribute description of the type:

$$V = \Sigma\, w_i V_i \,,$$

where V_i are the different, normalised, vulnerability indicators and w_i are the corresponding weights. The weights are related to the importance of the different vulnerability indicators. These concepts will be explained in more detail in the example in Section 4.2.

In the context of safeguards, three vulnerability indicators are proposed:

1. *Detection probability*: probability to detect a diversion or undeclared activity during one inspection. A higher value of the detection probability corresponds to a lesser vulnerability.
2. *False alarm probability*: probability that an alarm will be generated when no diversion or undeclared activity has taken place. A higher value of the false alarm probability corresponds to a higher vulnerability.

3. *Delay time*: time between the moment a diversion or undeclared activity has taken place and the moment of detection. A higher value of the delay time corresponds to an increased vulnerability.

In case of more than one diversion scenario (as it is the case for both the open and the closed repository), vulnerability has to be evaluated for each scenario separately. One can subsequently evaluate which safeguards measures are effective for all possible scenarios. Although cost is not considered in this paper, it is certainly an important factor that may influence the choice of one or another (combination of) safeguards measure. For instance, one may seek to select a strategy that is sufficiently effective for all scenarios (although it might not be optimal for all of them) and that is also feasible with respect to the implementation costs.

Evaluating the vulnerability of a geological repository poses some challenges. At the outset of this study, it was intended to require from a number of experts to evaluate the three vulnerabilty indices -in terms of a best, worst and most likely value-, for a general model of repository and for all meaningful individual measures or combinations of measures. This however proved unfeasible, on the one hand due to the uncertainty of such evaluations, on the other hand because the effectiveness of the measures highly depends on the geological repository considered and the way the implementation of safeguards measures is envisaged (e.g. frequency of performing the verification, etc). Therefore, we present in the next section how this kind of analysis may be performed, but narrowing our scope to an illustrative example.

4.2 Illustrative example

In the following we exemplify a vulnerability assessment study for a closed repository.

Based on INFCIRC/153, the technical purpose of safeguards is: "…*the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection*."

Three experts in safeguards of geological repositories were asked to fill in a questionnaire in order to give an evaluation of above-mentioned three parameters for several combinations of safeguards measures for the different mentioned diversion paths. Due to time constraints we limited the evaluation to the closed repository.
Moreover, following assumptions regarding the carrying out of the safeguards measures were made by one or more of the experts:

- The State falls under Integrated Safeguards
- Frequencies of inspections and satellite monitoring were not specified in the questionnaire, therefore the experts made some assumptions based on present practice, like nuclear material accountancy takes place once per year plus two annual random inspections. Satellite monitoring once in three years. Design Information Verification is performed annually for an open repository and each five years for a closed repository.
- Passive seismic monitoring is performed continuously but analysed monthly.
- Time for re-opening shaft or creating new shaft is estimated differently by different experts.

We shall further analyze the first diversion path described in subsection 3.2, reopening of the original shaft. The different safeguards measures considered for a closed repository are presented in Table 1, whereas some expert estimations of the three vulnerability indicators for the selected diversion path are given in the Table 2.

Measure	**Short name**
Nuclear material accountancy	NMA
Design information verification: visual	DIVV
Seismic passive methods	SMP
Seismic active methods SMA	SMA
Satellite imagery	SI

Table 1: Safeguards measures for a closed repository

Measure	Detection probability (%)	False alarm probability (%)	Delay time (days)
NMA	0	0	inf
NMA, DIVV	75	0	180
NMA, DIVV, SMA	85	8	180
NMA, DIVV, SMP	80	3	180
NMA, DIVV, SMA, SMP	85	4	180
NMA, DIVV, SI	80	10	90
NMA, DIVV, SI, SMA	90	15	90
NMA, DIVV, SI, SMP	85	12	90
NMA, DIVV, SI, SMA, SMP	90	13	90

Table 2: Evaluation of vulnerability indicators for the selected diversion path, reopening of the orginal shaft

In order to carry out the vulnerability analysis for the selected diversion path we employed a multi-criteria decision-analysis tool called Web-HIPRE (Mustajoki [5], www.hipre.hut.fi). We first construct a hierarchical structure (also called value tree) starting from the main objective which is minimising vulnerability, i.e. maximising resistance to diversions. On the next level of the hierarchy we have the three vulnerability indicators and finally, on the last level we place the relevant (combinations of) safeguards measures, as illustrated in Figure 2. The first nine combinations of measures include always nuclear material accountancy, although not specifically mentioned in the figure.

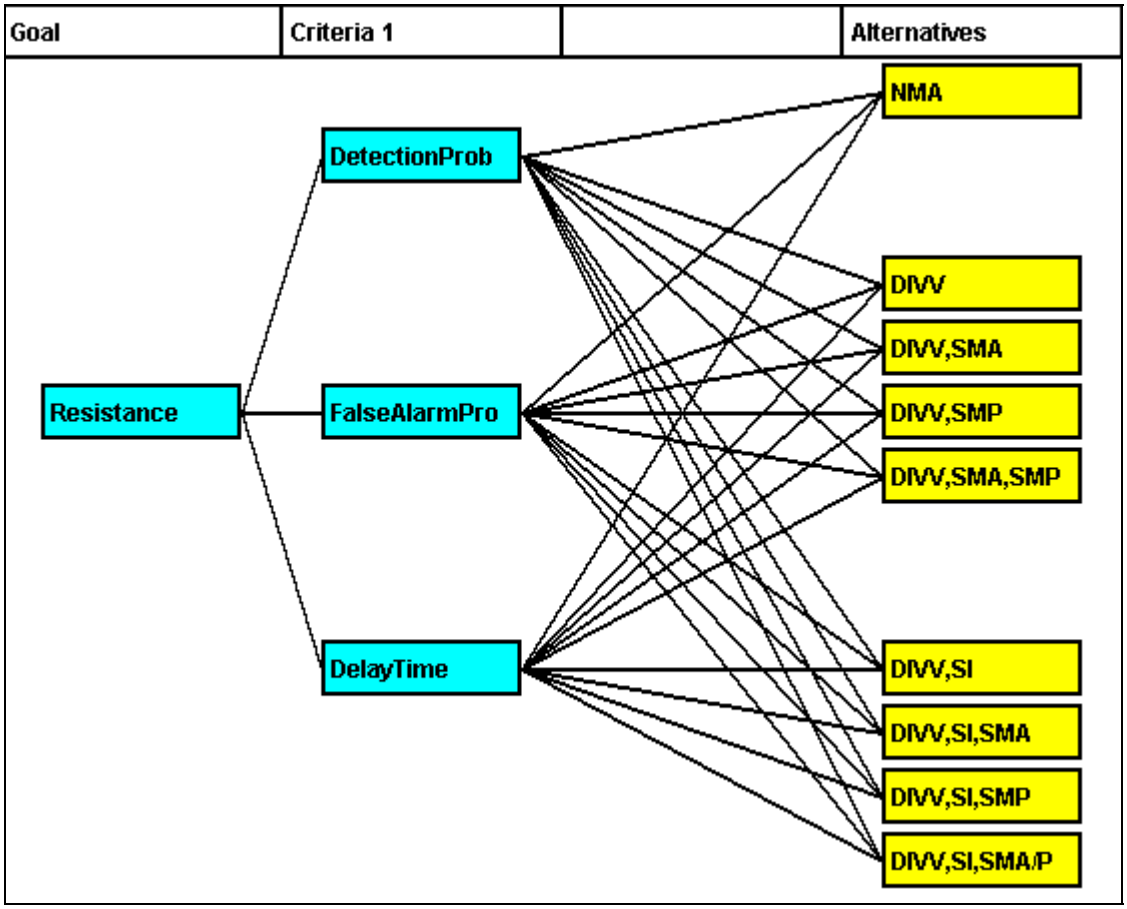
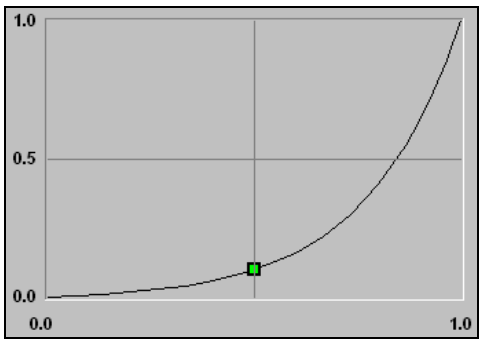
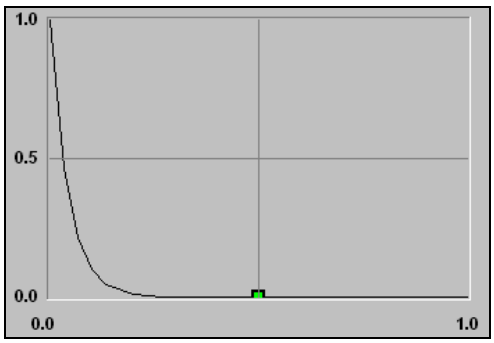


Figure 2: Vulnerability value tree showing the overall goal, the three vulnerability indicators and the combinations of safeguards measures

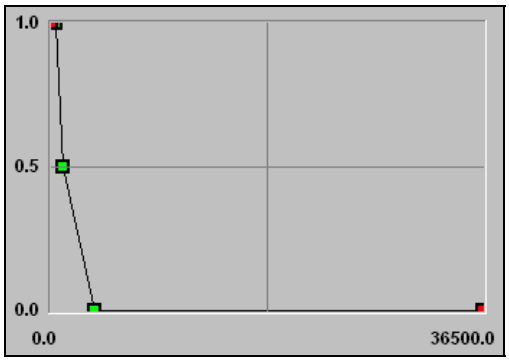
At the next step we construct value functions which describe the evaluation of the performance of the safeguards measures on the three vulnerability indicators, with respect to the overall goal. In other words, the evaluations of three indicators (in the respective units, e.g. days) will be translated in value scores in the domain [0, 1]. The value functions constructed are presented in Figure 3.



Value function for detection probability:
exponential with value(0.5)=0.1



Value function for false alarm probability:
exponential with value(0.5)=0



Value function for delay time:
piecewise linear with value(360 days)=1, value(1000 days)=0.5, value(3700)=0.

Figure 3: Value functions for detection probability, false alarm probability and delay time.

The value function for detection probability has been constructed based on the consideration that there is a relatively strong preference for high detection probabilities. Detection probabilities below 50% were given a low value (less than 10%), while only for detection probabilities higher than 80% the value function becomes higher than 50%.
The value function for the false alarm probability has been constructed based on the consideration that medium and high false alarm rates are not tolerable for safeguards inspections. These false alarm rates would undermine the confidence in the safeguards inspections and would increase the costs too much due to the necessity to repeat often (parts of) inspections.
The value function for the delay time is based on the considerations that: a delay time of 1 year is perfectly acceptable, certainly when taking into consideration that the reopening of the original shaft could take easily 2 years; in view of the previously mentioned reopening time of 2 years, a delay time of 3 years may still be acceptable; a delay time of more than 10 years is certainly not acceptable for safeguards.

It is clear that the value functions presented here have been constructed on partly rather arbitrary considerations, but the general safeguards philosophy is kept, although details of the value functions may be adapted following more profound discussions.

The weights for the different indicators have been derived with the AHP method (Analytical Hierarchical Process by Saaty [6], based on pairwise comparisons of the three indicators, with the help of the Web-HIPRE software. Detection of a diversion has the highest relevance for safeguards since without the detection of a diversion the main safeguards goal is not obtained. Delay time has been considered more relevant than false alarm probability, since the delay time is directly connected to "timely detection", whereas a higher false alarm probability will cause some reverification activities and has therefore mainly budgettary implications, assuming reverification activities can be performed appropriately. The derived weights were 0.714 for the detection probability, 0.223 for the delay time and 0.063 for the false alarm probabibility.

The results of the assessment are summarised in the matrix below (Table 3), giving the weights for the various combinations of safeguards measures for the mentioned diversion scenario.

Values scores	NMA	DIVV	DIVV, SMA	DIVV, SMP	DIVV, SMA,SMP	DIVV, SI	DIVV, SI,SMA	DIVV, SI,SMP	DIVV,SI, SMA/P
detection probability	0.0	0.232	0.365	0.291	0.365	0.291	0.457	0.365	0.457
delay time	0.0	0.223	0.223	0.223	0.223	0.223	0.223	0.223	0.223
false alarm probability	0.063	0.063	0.010	0.032	0.025	0.006	0.002	0.004	0.003
overall vulnerability	0.063	0.518	0.598	0.546	0.613	0.520	0.682	0.592	0.683

Table 3: Summary of results for the vulnerability assessment in the example considered

5. Discussion

The vulnerability values for the delay time are the same except for NMA only. This is due to the fact that the value function has a value of 1 for a delay time smaller than 1 year, which is the case for all combinations except for NMA. Therefore the delay time hardly plays any role in making a distinction between the different combinations.
Due to its low weight, the false alarm probability has a low influence on the overall vulnerability of the different combinations.

Several experts were asked to give their estimates of the best, worst and most likely values of the three indicators detection probability, false alarm probability and delay time. However, this proved to be unfeasible and the received results were difficult to compare, since they were based on different assumptions regarding the implementation of the safeguards measures. Therefore the methodology is given in this paper, applied to a limited set of data received from the experts. Parameters that could not be evaluated are the uncertainty of the experts' opinions, based on their estimates of best and worst values and possible differences in opinion of the experts regarding combinations of safeguards measures.

Based on above-mentioned example, the best combination of safeguards measures is the combination that comprises all considered methods. As such this result is not surprising, but the very small difference of the resistance value with the next best combination (NMA, DIVV, SI, SMA) indicates already that further analysis, taking into account e.g. costs, will reveal the usefulness of the applied method.

An analysis of multiple scenarios will as such not change the best possible combination of safeguards measures, but will reveal the best order of combinations based on all diversion scenarios. For safeguards it is very relevant that all diversion scenarios are covered by the safeguards approach in an equal way. As mentioned in section 4.2., diversion scenarios are not assessed regarding their likelihood, based on e.g. the cost of a diversion. Would this be the case, a diverter would probably follow the most expensive diversion path in order to reduce the detection probability.

It is intended to show the applicability of this method with the example of safeguarding a geological repository. There are no restraints for applying this method to other safeguarded installations.

6. Conclusions

Further work will focus on the gathering of additional data from multiple experts by defining the boundary conditions of the assessment in a clear way. This will clear the way to analyse multiple scenario cases, taking into account uncertainties of the evaluation process.

An important additional factor that should be taken into consideration is the cost of the combination of safeguards measures. In this way the optimum combination can be evaluated and the price per additional % increase of detection probability can be assessed.

Therefore a multiple scenario analysis, combined with incorporation of the costs of the safeguards measures and a profound discussion with the stakeholders about the value functions, will result in valuable knowledge about the most efficient and effective combination of safeguards measures for geological repositories.

7. Acknowledgements

The authors would like to acknowledge Bruce Moran from the IAEA, Olli Okko from STUK and Per Grahn from SKB for their valuable contribution to this paper.

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Integrated Safeguards for the geological repository in Finland

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Abstract:

As a result of the early decision to focus on geological disposal of nuclear fuel, geological site investigations have been carried out in Finland since the early 1980s by the nuclear power companies in order to identify a safe site for a repository. After acceptance by the local municipality and Finland's Council of State, the plan for final disposal of spent nuclear fuel in the geological repository at Olkiluoto in western Finland was finally endorsed by Parliament in 2001. The present plan is to construct an "Underground Rock Characterisation Facility" called Onkalo, consisting of an access ramp and a system of tunnels and shafts. It is expected that the first underground access tunnel will be part of the final disposal facility and nuclear material may be moved through the Onkalo tunnels to be emplaced in the bedrock. The National Nuclear Non-Proliferation System has been effective from the beginning of the excavation phase in order to enable continuous design information verification and assurance on the absence of undeclared safeguards relevant activities and to make possible future safeguards activities by the IAEA and the European Commission. These site-specific activities and their timelines are to be adjusted in accordance with the general Integrated Safeguards framework in Finland.

Keywords: final disposal of spent fuel, geological repository, continuity of knowledge, integrated safeguards

1. Introduction

The decision to construct an underground repository for spent nuclear fuel initiated the development of new types of geoscientific site characterisation methods in the 1980s. The objective is to identify rock volumes sufficient for the building of a repository that fulfils public long term safety requirements and will protect future generations from the harmful effects of radiation. Based on information from geological site investigations and after a parliamentary decision of 2001, the final spent nuclear fuel repository is to be located at the Olkiluoto site in Eurajoki, western Finland. The current phase of site investigations includes the construction of underground premises for rock characterisation purposes. The excavation of these galleries for the Underground Rock Characterisation Facility Onkalo began in 2004. At present (May 2009) more than three kilometres of entrance ramp extend to a depth of 350 m below ground level. After the investigation phase, nuclear licensing is scheduled to take place by 2012.

The final disposal of the nuclear material will introduce new safeguards approaches which have not been applied previously in the International Atomic Energy Agency's (IAEA's) safeguards for spent fuel. The encapsulation plant to be built at the site will be the final opportunity for verification of spent fuel assemblies prior to their transfer to the geological repository. Moreover, additional safety and safeguards measures are being considered for the underground repository [1]. Safeguards verification systems will be based on the design information verification (DIV) of the declared excavated rock volumes, already during the construction phase. Later, Nuclear Material Accountancy will remain the fundamental safeguards measure and Physical Inventory Verifications (PIV) will be performed accordingly. In addition, geophysical monitoring may be applied to support containment and

surveillance to detect unannounced activities related to possible reprocessing or diversion schemes of nuclear materials at the repository site.

Onkalo is not yet a facility subject to traditional safeguards since no nuclear materials are licensed to this “non-nuclear installation”. Similarly, the repository site or the geological investigation site does not constitute a site delimited under the Additional Protocol (INFCIRC 540). Basic information on the facility will be provided to the Commission at least 200 days prior to the beginning of construction of the nuclear facility. The disposal site, which will consist of the encapsulation plant and the underground repository, will be delineated by the State. The Basic Technical Characteristics (BTC) will be provided to the Commission at least 200 days prior to the arrival of the first batch of nuclear material. Based on these, the Commission will prepare the Design Information (DI) for the IAEA. In the meantime, Onkalo has been declared by Finland as falling within the general plans relevant to the development of the nuclear fuel cycle, under article 2a(x) of the Additional Protocol. Formal DI Questionnaires and BTCs - recently been drafted by the IAEA and EC for encapsulation plants and geological repositories respectively - have not yet been considered for the new type of facility. A joint field trial has been initiated to obtain feedback as to the usefulness and relevance of these documents and to start the formal implementation of safeguards. The experience gained from the planning of containment and surveillance measures for new nuclear installations under construction (see [2] for the disposal facility) clearly shows the need for early provision of even preliminary design information for safeguards purposes.

2. New challenges

Referring to the recommendations generated in the IAEA’s Programme for Development of Safeguards for Final Disposal of Spent Fuel in Geological Repositories [1, SAGOR phase, continuation is referred to as ASTOR, Application of Safeguards to Repositories], STUK established (in 2002) and runs the National Competence Network [3, 4] involving the implementing company and other relevant stakeholders, in order to fulfil the State’s obligations under the Non-Proliferation Treaty of Nuclear Weapons even at this early stage in the development of the repository. The National Non-Proliferation System is to generate the safeguards-relevant database needed for BTC and DI purposes at the time of licensing and commissioning the nuclear facility. Currently co-operative negotiations are continuing under the SAGOR/ASTOR framework between the Finnish and Swedish State representatives (STUK and SSM), EC and the IAEA to define the international current data and information requirements and to establish an inspection framework during the present pre-nuclear phase of repository development.

2.1 Baseline information & Continuity of Knowledge

The decision to construct an underground repository for spent nuclear fuel initiated development of site characterisation methodologies and investigation programmes at several sites in Finland. After the selection of the Olkiluoto investigation site to be the disposal site for the spent nuclear fuel in Finland these methods were tailored to the site-specific geology [5]. Before the underground excavations of Onkalo started, the hydrogeological, hydrogeochemical, rock mechanical, tectonic and seismic conditions were documented in a baseline report to serve as a reference point [6]. These characteristic were considered relevant to the long-term safety of the repository to be constructed. These were evaluated for their safeguards relevance during the SAGOR II phase [7]. Since some changes or fluctuations in these characteristics are expected due to underground construction work, a monitoring program [8] is planned and scheduled to fit in the activity plan. The monitoring programme is reported annually to provide continuous documentation.

During the construction of Onkalo, next to site characterisation the main focus has been on the generation of credible regulations for documenting construction and adjoining area geoscientific monitoring records to survive over the 100-year disposal project. The timely documentation of the planned [9] and, in particular, the excavated [10], and later back-filled rock volumes is intended to generate the Design Information declarations to be verified as safeguards measures during the operational time of the repository. The current DIV technologies applied to safeguard the repository are based on the standard surveying used as a part of the rock characterisation process. After the closure of the repository these documents will also serve for the inventory maps providing the Continuity of Knowledge, CoK, for future generations.

2.2 Site-specific novel technologies

The application of geophysical exploration methods as internationally accepted verification practices points to the need to have accepted standards and proven technologies to fulfil the regulatory requirements. The monitoring program should bring in all the necessary data to make safety-related observations if unexpected phenomena are located in the geological formation near the repository. It is therefore proposed that possible observed instances of safety reduction be used as safeguards alarms [11]. Although the interpretation of independent remote sensing or geophysical sounding is always a result of subjective survey planning, performance and reporting - and thus open to dispute - the reliability level of the methods applied should satisfy the threshold level required for conclusive safeguards verification measures. Owing to their nature and the indirect interpretation procedures, the indirect methods can be treated as scientific reports, but not as declarations to be verified. The site-specific monitoring programme is reviewed for its safeguards relevance to enable the IAEA to plan site monitoring [12]. The current national requirement is to analyse the seismic records not only for the long-term safety analysis for which they were collected but also for their safeguards-relevance. The presence of the remote sensing instruments is supposed to have a deterrent effect to prevent unexpected intrusive accesses to the repository.

2.3 Integrated safeguards approach for the Finnish repository

The implementation of Integrated safeguards in Finland began on 15 October 2008. According the current IAEA approach one PIV/DIV and a few Random Interim Inspections will be carried out annually at the NPPs. The PIVs at minor facilities, including the research reactor, two universities and STUK will be performed at 4-6 years intervals. In addition, the IAEA may perform complementary accesses, short notice random inspections and unannounced inspections at the Finnish facilities as agreed. This has been considered sufficient for the Finnish nuclear fuel cycle that consists of the fuel for nuclear energy power production at the NPPs, supported by education and research activities. There are no plans for the reprocessing of spent fuel. The spent fuel assemblies will be encapsulated as they are and disposed of.

There are two new facilities under construction in Olkiluoto, the new power reactor and the repository. The implementation of the traditional safeguards approaches requires early provision of even preliminary Design Information in order to start negotiations about the needs for safeguards instrumentation in time. Design Information Verification is expected to take place according to the construction schemes of the facilities. The construction of these new facilities does not have much of an effect on the Finnish fuel cycle and the integrated safeguards approach before the emplacement of spent nuclear fuel in the underground repository, instead of above ground storage. Annual PIV/DIV timelines at the repository shall be adjusted according to the general Integrated Safeguards framework in Finland. The confidence and CoK about disposed material must be maintained at state level because there are no means to verify the material in the repository.

3. Summary

The final disposal of the nuclear material will bring with it new safeguards approaches which have not been applied previously in the IAEA's safeguards for spent fuel. The long life time from investigations, research, construction, operation and to its final closure has to be given careful consideration. The present stage of underground construction can be verified through human access and by use of geodetic underground surveying techniques. This requires continuous presence at the installation. The cost-effective and non-intrusive national safeguards system is based on the timely validation and verification of the underground geological site investigation process. The procedures to carry out DIVs (and later PIVs) are the subject of the current IAEA-EC-State field trial under the SAGOR/ASTOR framework.

In the future repository, there will be no direct methods to verify the disposed nuclear fuel. Therefore, the safeguards approach will be based on measures that will effectively utilise information and data from the safety-related confidence-building processes. The integrated safeguards approach for the state as a whole is well fitted for this purpose.

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Spent fuel verification and continuity of knowledge through centuries

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Abstract:

NDA-measurements are an important and widely recognised tool in the safeguards arsenal. However, the wealth of data produced by these measurements often goes largely to waste. While the results are valuable when analysed singly, it is possible to gain better verification certainty from the measurements if the historical data is also taken into account when conducting the analysis. Taking advantage of these possibilities requires a well-designed data storage system so that the historic data can be easily and effectively used.

The nature and amount of the data that has to be stored for the historic analysis requires makes a database-based system a natural candidate for the basis of data storage. However, the special characteristics of NDA-applications, especially the long storage times and the number of different verification techniques, create unique challenges in the design of the system. Thus, the design of the data storage system must start with a clear understanding of the requirements of the NDA-applications.

A further challenge for the system is presented by the final disposal of the fuel assemblies. As most of the fission products will die out during the expected interim storage, verification of the contents of the assemblies just prior to the final disposal can be very difficult and time consuming. Comprehensive storage of verification measurement results offers a partial solution for this problem. The full verification history gives a picture of the content of the assembly during the interim storage. Thus, if the measurement history is available through the database, and the integrity of the assembly can be verified, the contents of the assembly can be ascertained.

The use of the measurement results does not end at final disposal of the assemblies, either. After the final disposal, the old verification measurement results are the most accessible source of answers for any question about the assemblies that have been placed in the final repository. Thus, the database system should be designed with the assumption that the information must be accessible for extremely long times.

Keywords: NDA, databases, data management

1. Introduction

NDA-measurements are widely used by both international and national agencies responsible for the safeguards verification of nuclear materials. Yet, practically all of the measurement strategies and research in the area are done with the assumption that every measurement is done without any knowledge of the previous measurements. While this must obviously be the case when measuring assemblies for the first time, the later measurements of the same assembly, possibly even several years later, can result in better, more reliable verification results. Especially in techniques that rely on operator-declared burnup, the previous measurement in effect takes same kind of a role as the burnup has in previous measurements.

Taking advantage of the earlier measurements naturally requires that at least the analysis

results, preferably the whole measurement results, are stored and available for future reference. This requirement makes the same kind of storage system good basis for accounting fuel assemblies moved into final disposal. The NDA-measurements and analyses form the basis of information the would be required in such a system as these kind of measurements offer the best verification of the identity of the assemblies that is possible without digging the assemblies up from the repository. In addition, the final disposal does require storing some extra information about handling and location. However, the biggest additional demand from this use comes from the information assurance demands. The stored information must still be available, understandable, and authentic perhaps even centuries forward. This does place some additional demands and points of emphasis on the design of the system. In practice this mean that the system must be based on some kind of database structure.

2. Improving analysis results through data storage

The various NDA-techniques in use and in development are result of large amount of research and technical characterisation, both theoretical and practical. However, the starting assumption in these reports is practically always that the fuel assembly, or other object of interest, is being measured for the first time. While this kind of measurement is essential for verification purposes, it is not the only setting for these measurements - perhaps even not the most common one. For example, a strategy for verifying spent fuel in a power plant might call for measuring 40 assemblies twice a year while the plant would produce around 50 spent assemblies per year. Obviously, this would mean that several of assemblies would always be assemblies that have been measured earlier. Even if the number of measured assemblies is less than the production of “new” spent assemblies, measurements should not concentrate exclusively on previously measured assemblies to obtain reasonable assurances of non-verification. Thus, many verification measurement strategies dictate that at least some of the measurements are done assemblies that have been measured earlier.

Most, if not all, organisations conducting verification measurements have some kind of database or similar storage location for the results. However, much like the research on NDA-techniques, typical starting point while designing this kind of database are thinking the data as single, one-off measurements. As a result, the database structure becomes easily such that it really support only “one-way” data transfer: the measurement and analysis results are stored into the database and then left there unless something unexpected development requires reviewing the older results. Yet, the data from previous measurements can allow better analysis results if the system is integrated into the analysis project more closely, making the data available easiler in a usable format during the analysis - and preferably fetched and inserted automatically into the analysis. A well-designed system also allows use of earlier measurement results as much as applicable even if they result from different measurement technique.

To illustrate some of the possibilities, two examples are presented in following chapters. First one presents a situation where an assembly is measured during several measurement campaigns with a relatively simple GBUV-method. Second one presents a situation where an assembly is measured with two different methods (eFORK and SFAT) and how the measurements can be used to support each other. Both of these examples use quite simple comparisons with earlier results and simple additions to standard analysis software. A more through mathematical analysis and/or more sophisticated modifications on analysis software would probably allow even better improvements using the data of older analyses.

2.1. Repeated measurements with same method

As the first example, we consider two measurement campaigns conducted with GBUV-method [1], which uses gamma spectrometric analysis of fission products (especially Cs-137 and Cs-134) to determine the relative burnup of measured assemblies. The analysis results in relative burnups, which are compared to the operator declared burnups. This relativity and reliance on operator-declared information mean that the GBUV-method can easily gain advantage from the analysis database.

In this case, the first measurement of an example assembly is done when the assembly's cooling time is still relatively short. This analysis is done as with a one-off measurement, yielding a relative burnup result. The analysis of all of the measured assemblies results also in a fitted curve that describes the relation of these relative results to all of the operator-given burnups, example of the result is presented in figure 1. Thus, we have an estimate for the true burnup of the assembly for later measurements - and as this derived using same or similar measurement technique as is used in later campaigns, the later results are expected to match this value better than the operator-declared value, which is calculated with some approximations and may have 5% (or even larger) uncertainties. While this does not increase the trustworthiness of this particular result for example intentionally misreported burnup, it is an important base result for future. Assuming that the cooling time is indeed such that the shorter-lived nuclides are still detectable, these can give some assurances that the assembly has not been tampered with.

The advantages of the database storage become apparent during later measurements. Simple comparison of the results itself already can give a very good certainty that the assembly hasn't been tampered with between the measurements as the measurements match closer to each other than the declared burnup. Even better comparison can be made if the trendlines of measured signal against the burnup are matched before the comparison, as this should remove the measurement setup specific variance. Again, in practice, this should be as automated as possible to not to interfere with the measurement process. Furthermore, as the shorter-lived nuclides are most likely barely detectable, if at all. So, the ability to connect the new measurement to one where they are still available might help with the verification and the analysis.

Granted, all of these simple examples are doable with hand analysis if the older data is available. However, a solution built around a database allows automatic and fast implementation of this without the need of human intervention. The main advantage obtained that is hard to replicate with non-integrated setup is that the setup doesn't need a dedicated calibration assembly to calibrate results from different measurement campaigns. If the pool of possible assemblies to be measured stays the same, it is possible to select few assemblies to be remeasured in a later campaign, resulting in enough datapoints to give a good spread of reliable calibration between the campaign results. This can be obtained with a relatively few assemblies, so it has minimal effect on the total time required for the measurement campaign.

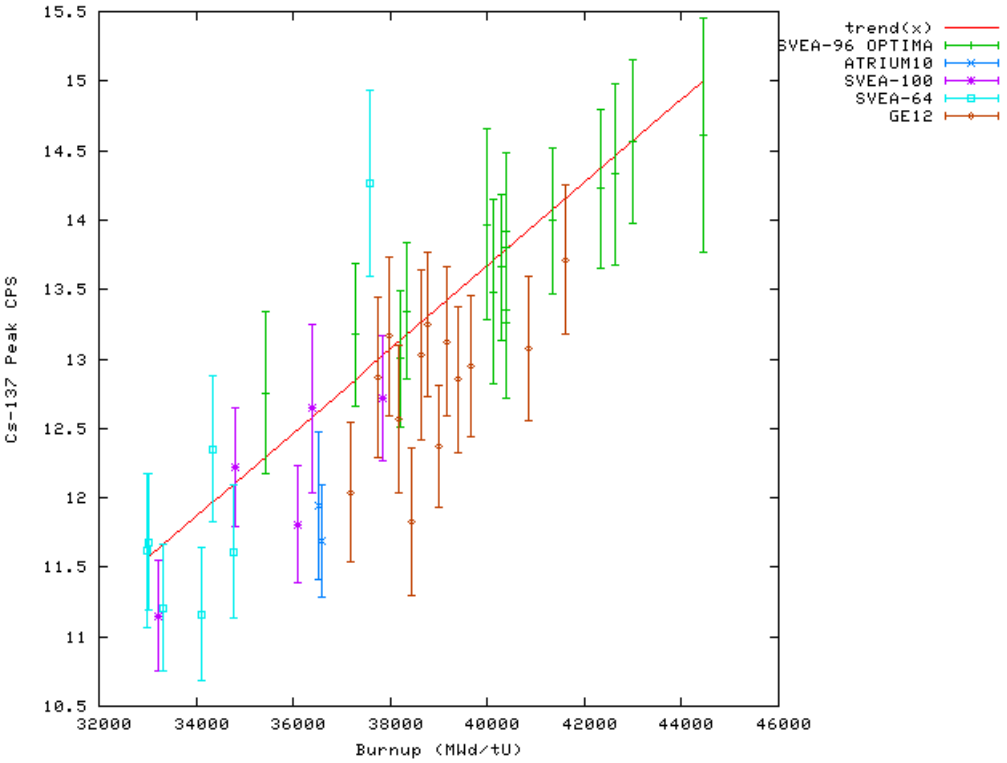

Figure 1. Example plot of measured assemblies from measurement campaign and trendline based on older results. This view is plotted directly from database and is available online during measurements.,

2.2 Measurements with different methods

As a second example, we take situation where two different measurement devices are used to verify the same fuel assembly in different campaigns. In this example, the methods used are eFORK [2] and SFAT [3]. The eFORK is a more complex device, having gross gamma, neutron, and gamma spectroscopic detectors. With these measurements the eFORK device can be used to assess the burnup of the assembly independent of operator declared values. The neutron measurements are also more sensitive to changes in the centre rods of the assembly than gamma measurements. Thus, the verification results from eFORK have a very good quality. However, the measurement process is relatively slow, as it requires long moves of the spent fuel. SFAT, on the other hand is a simple system that takes a gamma spectroscopic measurement from above the assembly. This type of measurement is much faster to conduct than the eFORK measurements, but can only see the top of each assembly.

When both of these types of measurements are stored in the same database, it is possible to verify their consistency relatively easily. Using different kinds of measurement techniques to verify the same set of assemblies is thus possible without losing the advantage of comparable results that using single type of measurement has. This makes it possible to always select the most optimal verification measurement for the planned time and desired verification accuracy without need to consider the techniques used in the previous measurement on the same set of assemblies.

3. Challenges from final disposal

Final disposal of the spent fuel presents additional challenges for the verification and data storage. When considering these, the problems can be conceptually divided to three questions: What do we think we are placing in the repository? Is it really what we think? How can we ascertain this later if needed?

The first question falls mostly in the realm of nuclear materials accounting; the measurement-centred data storage under discussion here has only incidental use with that question. However, the data storage is much more useful when considering the second question. The plans for the disposal contain verification measurement of all of the assemblies to be placed in the final disposal just before the actual disposal takes place. There are several techniques being considered for these verification measurements, such as gamma tomography [4] to obtain more detailed GBUV-like picture of the assembly, or a combination of few different simultaneous techniques to directly measure the Pu-content [5]. In the first type of technique earlier measurement results are very useful as a baseline comparison to ascertain that the assembly being measured is indeed what is expected. Thus, the situation is very much special case of the situation describes in chapter 2.1 above. The second approach on its part requires combination and storage different types of measurement results. That, in turn, is similar to situation described in chapter 2.2. Thus, both approaches for verification of the fuel to be place in the disposal gain the same advantage from data storage structure as “regular” verification measurements.

The third question phrased above is the question that makes the database-based storage for the measurement results essential. The safeguards interest on the spent fuel assemblies does not end in the moment of final disposal, but the access to it becomes hard or impossible, prompting some extra preparations. One part of this is, of course, assuring that the assemblies are not diverted away from the repository. However, that part is not part of this scope of this paper. The other part is ensuring that possible questions of the identity of the assemblies can be answered later. Obviously, digging up the assemblies to get answers about the composition of the assemblies is not a practical solution, so a data storage that contains the measurement results from the verification campaigns can offer a reasonable solution for this. If the measurement results and parameters be available in adequate detail in the database, it should contain enough information to ensure that the questions about the material can be answered and that all of the material can be identified later as well.

This level of interest on the material placed in the final disposal can be expected to continue

for at least as long as there is more material placed in the repository, most likely considerably longer, even several centuries. This means that there is a considerable demands placed on the database: the data must be readable, understandable, and authenticable even after long times. In practice, it means that there has to be either a readiness to devote resources to further development of the system as computers and software moves forward, or the system has to be based on software solution which can be expected to have updates available. As the analysis methods can be expected to improve as well, the database should contain the raw data from the measurements as well as the analysis results themselves to facilitate reanalysis of the measurements with newer analysis techniques, if desired.

4. The Finnish system

Both to support the ongoing verification campaigns and to prepare for the final disposal of the spent fuel, the Nuclear Materials Office of STUK has developed a database system for the verification measurements. The database is also expected to fulfil the role of storing the measurement results after final disposal begins at the geological repository in Olkiluoto. The system is based on Linssi-database [6], which is a free MySQL-based database developed for radiation measurements, especially gamma spectroscopy. However, the database can support other types of measurements without modification. Only modification required for the verification use is addition of one table that contains information specific to fuel assemblies (burnup, reactor periods, and so on). As a further development of originally Finnish CTBT NDC database, it has been designed from ground up for large amount of data and storing all of the raw data obtainable from the measurements - two requirements also present in the verification use.

One of the main advantages of using existing system as the base for this purpose is that there are several ready tools available. This cuts down the need to devote development time for the common parts of the system, such as GUI for browsing the stored results or administration tools. In addition, it means that upgrades for the system can be expected to be easily available in future. The Linssi system, in addition, is already supported by several measurement programs, including UniSampo, which is the main software used with verification measurements in Finland.

The Finnish setup also takes advantage of the defined transfer protocols that allow transferring data between two different databases easily. The setup, presented in picture 2, has the central database, which contains all of the measurement results and is used as the main data storage. In addition to this, all of the laptops used in verification measurements have a version of the same database installed. This copy of the database is updated before each campaign to contain the most up-to-date information and previous measurement results of the assemblies that are planned to be measured. This allows online analysis and decay and other corrections during the measurements, as well as immediate comparison of the results with older results. New measurement results are inserted to this database and then copied back to the main database at the main office.

The database is expected to be the basis for the future data storage of fuel in geological repository in future. The database can already include all of the required information for this purpose. Thus, the main effort in future will be to keep the structure and software of the database up to date with the progress of software technology.

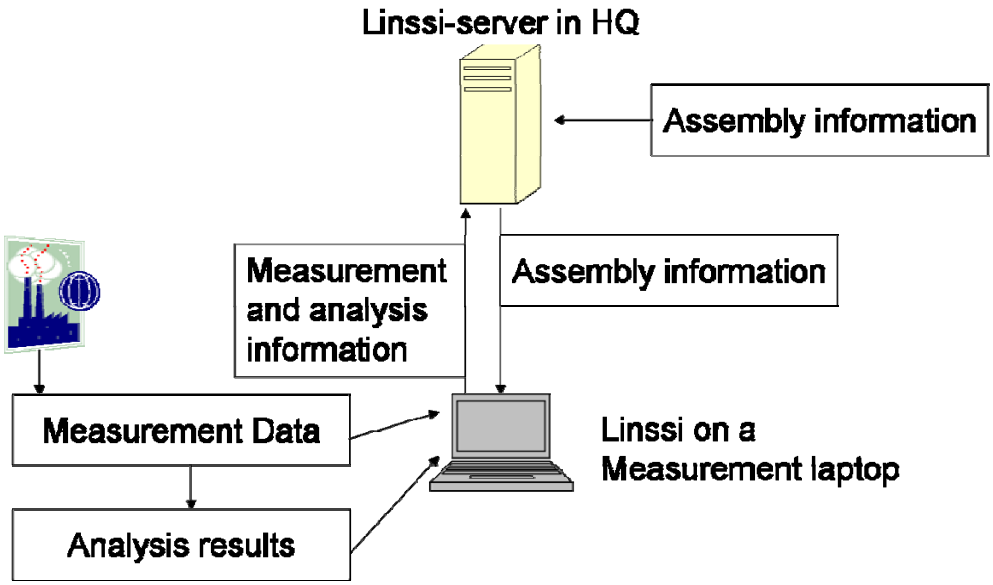

Figure 2. Conceptual view of data flows when using central Linssi server together with light measurement version.

5. Conclusions

A database structure for storing information about spent fuel being placed in the final repository is critical for guaranteeing the continuity of knowledge for required time. However, the long storage times require some additional considerations. Namely, the structure must be easily applicable for upgrades and the data must be comprehensive enough to answer all of the possible questions without relying on outside sources.

Implementing the database already during routine verification measurements allows populating the database with information well in advance of the final disposal. In addition, well-integrated measurement database makes helps in quality and speed of verification analyses, yielding advantages well before the disposal.

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Uranium mass measurements in waste drum by gamma spectrometry: an evaluation of the measurement uncertainty

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Abstract

A technique is developed at the "Institut de Radioprotection et de Sûreté Nucléaire" for measuring quantity of plutonium contained in waste drums and has been adapted to the measurement of uranium mass, taking advantages of the capabilities for the new code IGA to analyse complex spectra. The method is based on a gamma spectrometric acquisition device and spectra are analyzed using the so called « infinite energy extrapolation method ». It uses several energy peaks of U-235 and U-238 to evaluate the self-absorption correction of the uranium in the waste drums. This paper presents the method of measuring the mass of uranium in waste drums by γ spectrometry using the home-made MaPU software, the factors that influence the measured result, the calculation of uncertainty and the performance of the measuring instruments.

Keywords: *Gamma spectrometry, uranium waste, uncertainty*

1. Introduction

The IRSN has to carry out non destructive assay of nuclear material as part of its mission of technical assistance to the Authority responsible for the protection and control of nuclear material in France. In order to carry out these assays properly, IRSN has developed ways of making quantitative and qualitative measurements of plutonium and uranium.

For measuring quantities of nuclear material in waste drums by gamma spectrometry, IRSN uses two calculation codes:

- PLUM which interfaces with the PUMA [1] code for plutonium.
- MASSU for uranium.

In using these methods, the following limitations have become apparent. Firstly, using the PLUM (PUMA) code, it is not currently possible to characterise "polluted" plutonium. This is because the peaks of some pollutants such as americium 243, neptunium 239, caesium 137, caesium 134, curium 244, curium 243, etc. interfere with plutonium peaks during gamma spectrometry. Moreover, IRSN does not have the technical means to modify the calculation code, since the PUMA code sources are not available. Secondly, the MASSU code does not allow peak deconvolution and needs an independent calculation code for the isotopic composition.

Given these limitations, IRSN has developed the MaPU calculation code, which means that it can:

- Use the calculation code of the isotopic composition IGA [2] and notably its analysis and peak deconvolution software for gamma spectrometry peaks;
- Calculate the apparent masses of uranium from apparent activities measured at different energies;
- Use the results from IRSN's "Scanning γ" device to determine the attenuation coefficients that need to be applied to a waste drum to correct the matrix effects on apparent masses calculated from measured activities [3];
- Calculate the mass of uranium by extrapolation to infinite energy.

MaPU is also designed for calculating the mass of plutonium or uranium in the form of "noble material", e.g. PuO_2 in an AA203 container.

MaPU is encapsulated in the automatic gamma spectrometry acquisition and analysis code AutoISO_PLUM [4].

At present the MaPU code provides:

- Wider scope of validity for PLUM, making it possible to apply PLUM to plutonium measurement even when pollutants are present;
- Improved reliability of peak deconvolution used for uranium;
- The possibility of characterising "noble material" (uranium and plutonium) with the same device.

This article presents the method of measuring the mass of uranium in waste drums by γ spectrometry using MaPU, the factors that influence the measured result, the calculation of uncertainty and the performance of the measuring instruments.

2. Measurement principles

2.1. Relation between counting and mass

The principle of measuring the mass of uranium in waste drums is based on the method developed by the French Atomic Energy Commission (CEA), known as "extrapolation to infinite energy" [1]. The PUMA code is based on this method. IRSN adapted this measurement method to analysing uranium spectra. It consists of a high resolution gamma spectrometer (germanium detector) that analyses the main gamma peaks emitted by the uranium between 140 and 1000 keV.

In a uranium sample, the mass M of uranium is related to the net count rate $N_{(E)}$ for the total energy absorption peak E by the equation:

$$M = \frac{N_{(E)} - B_{(E)}}{\varepsilon_{(E)} \times K_{(E)} \times P_{\gamma(E)}} \times C_{geo} \times C_{pile-up} \times F_{(x)} \times C_{att(E)} \times C_{aut(E)} \times C_{U(E)}$$

- $B_{(E)}$ is the net count rate excluding background noise.
- $\varepsilon_{(E)}$ is the total absorption efficiency at energy E for a distance X_0.
- $K_{(E)}$ is the ratio between the mass of ^{235}U (or ^{238}U depending on the energy level) and the total mass of uranium. $K_{(E)}$ is obtained by previous determination of the isotopic composition.
- $P_{\gamma(E)}$ is the photon emission rate per mass unit, i.e. the number of gamma photons of energy E emitted per second and per gram of isotope ($\gamma.s^{-1}.g^{-1}$).
- C_{geo} is the correction for changing from the calibration geometry, where the calibration sample is considered to be a point source, to the real sample geometry.
- $C_{pile-up}$ is the correction for count losses due to pile-up, i.e. the overlapping of separate impulses emitted within a time shorter than the electronic resolution time of the spectrometry chain.
- $F_{(x)}$ is the function transferring the efficiency response curve for a given distance X_0, to a different distance X. This is only used if the ratio of distances is under 3 [when $X = X_0$, $F_{(x)} = 1$].
- $C_{att(E)}$ is the attenuation correction due to the presence of screens between the γ emitter and the detector.
- $C_{aut(E)}$ is the self-absorption correction due to the sample, except for the uranium itself.
- $C_{U(E)}$ is the self-absorption due to the uranium alone.

The self-absorption factor associated with the $C_{aut(E)}$ matrix can be calculated either theoretically, or, for waste drums, after the transmission measurement. The self-absorption factor $C_{U(E)}$ is calculated differently, either theoretically for “noble nuclear material”, or using the extrapolation to infinite energy method described below for waste drums.

Once the corrections for geometry C_{geo}, transfer $F_{(x)}$, attenuation $C_{att(E)}$, and self-absorption $C_{aut(E)}$ have been made, the "apparent mass", $M_{app(E)}$ can be calculated using the following equation:

$$M_{app(E)} = \frac{N_{(E)} - B_{(E)}}{\varepsilon_{(E)} \times K_{(E)} \times P_{\gamma(E)}} \times C_{geo} \times C_{pile-up} \times F_{(x)} \times C_{att(E)} \times C_{aut(E)}$$

In theory, a different apparent mass is calculated for each energy level.

As for the real mass of the sample, this is independent of the energy level of the peak used to calculate it. The real mass is calculated from the apparent mass using the self-absorption coefficient $C_{U(E)}$.

$$M = M_{app(E)} \cdot C_{U(E)}$$

However, in practice, the coefficient $C_{U(E)}$ is never calculated. The mass of uranium is deduced from the apparent masses calculated at different energies by extrapolation to infinite energy, which can be given as:

$$M = \lim_{E \to \infty} M_{app(E)}$$

2.2. Application to uranium

Although the plutonium gamma emission spectrum has about a dozen peaks that can be used to plot the graph showing the relation between apparent mass and energy, a spectrum of uranium in waste drums acquired in working conditions (with limited acquisition time) has only six usable peaks, at energy levels 143, 163, 185, 205 keV for isotope ^{235}U, 766 and 1001 keV for isotope ^{238}U (or rather its decay product, or granddaughter, protactinium-234m which rapidly reaches equilibrium). These two very distinct groups of peaks, each group corresponding to one isotope, cannot - unlike plutonium - be used to test whether the isotopic composition used is valid and makes the measured result much more sensitive to variations in isotopic composition.

The ^{238}U peak at 258 keV is interesting because it provides a link between the two groups of high and low energy peaks. However, it is rarely used because it is not very strong and cannot generally be detected in the short measurement times required.

3. Materials and software used

MaPU uses the same γ spectrometry chain as PLUM. It is composed of a germanium detector, a digital analyzer and a plateau for aligning the detector and the waste drum and rotating the drum.

Detection efficiency is calculated using the EFFICACE code. This code uses the ETALON code developed by the CEA for PUMA.

Spectrum acquisition is carried out using the GammaVision software by AMETEK/ORTEC. Gammavision is piloted using the AutoISO_PLUM [4] code that enables incremental acquisition.

Isotopic composition is calculated using the IGA code. MaPU can also be used with an isotopic composition previously calculated by the various codes used at IRSN (MGA, MGA++, PC/FRAM) or entered directly by the user.

The net surface areas of the peaks used to calculate the apparent mass are those calculated using the IGA code (.sig file). In this way, the IGA deconvolution software separates out the multiplets. All nuclear data (decay periods, emission probability) come from the IGA code.

4. Evaluating measurement uncertainty

4.1. Method used

In order to evaluate the uncertainty of measurement associated with a mass of uranium using MaPU, the following error propagation formula is developed from the model described above:

$$\left(\frac{u_M}{M}\right)^2 = \left(\frac{u_{Compt}}{N_{(E)} - B_{(E)}}\right)^2 + \left(\frac{u_\varepsilon}{\varepsilon_{(E)}}\right)^2 + \left(\frac{u_K}{K_{(E)}}\right)^2 + \left(\frac{u_{P\gamma}}{P_{\gamma(E)}}\right)^2 + \left(\frac{u_{geo}}{C_{geo}}\right)^2 + \left(\frac{u_{pile-up}}{C_{pile-up}}\right)^2 + \left(\frac{u_F}{F_{(X)}}\right)^2 + \left(\frac{u_{att}}{C_{att(E)}}\right)^2 + \left(\frac{u_{aut}}{C_{aut(E)}}\right)^2 + \left(\frac{u_U}{C_{U(E)}}\right)^2$$

where

u_M	Uncertainty of uranium mass calculated by MaPU
u_{compt}	Uncertainty of net count rate
u_ε	Uncertainty of total absorption efficiency
u_K	Uncertainty of isotopic composition
$u_{P\gamma}$	Uncertainty of photon emission rate per mass unit
u_{geo}	Uncertainty of shape correction
$U_{pile-up}$	Uncertainty of count rate loss correction due to pile-up
u_F	Uncertainty of efficiency response curve transfer correction
u_{att}	Uncertainty of attenuation correction
u_{aut}	Uncertainty of matrix self-absorption correction
u_U	Uncertainty of extrapolation to infinite energy

Using this error propagation formula, the objective is to find the best possible estimate of the standard uncertainty for each term of the model input data.

In the present case, the relative uncertainties of the following factors are estimated by a type B method: u_ε, u_{geo}, $u_{pile-up}$, u_F, u_{att} and u_{aut}.

The relative uncertainties of factors u_K, u_{compt}, $u_{P\gamma}$ and u_U, are estimated by statistical analysis of a series of experiments, i.e. a type A method.

4.2. Uncertainties determined by a type B method

Total absorption efficiency

According to reference [6], the uncertainty of the total absorption efficiency for a temporary source can be estimated at between 1 and 3%. In the absence of a special study of the influence of efficiency on measurement, the reasonable value selected is $u_\varepsilon/\varepsilon_{(E)}$ **= 3**%.

Correction for geometry

For a 200 l drum measuring 57 cm in diameter by 80 cm high, using a detector placed 150 cm away (as recommended for measurements on waste drums), the radial and axial geometry corrections are approximately 0.965 and 0.934 respectively. The product of the two coefficients leads to a correction of approximately 10%. If we consider this to be the extreme value of the correction, and thus of the error committed, and using a normal error distribution law, the standard uncertainty is u_{geo}/C_{geo} **= 3%**.

Count loss due to pile-up

The overlapping of separate impulses emitted within a time shorter than the electronic resolution time of the spectrometry chain leads to an electronic dead time. According to reference [6], for a usual time constant value of 20 µs, count losses due to pile-up are approximately 20% for a count rate of 10^4 impulses per second. The relative uncertainty to use is $u_{pile\text{-}up}/C_{pile\text{-}up}$ **= 2**%.

Efficiency response curve transfer correction

Given the analytic expression of the transfer function $F_{(X)}$, the uncertainty of $F_{(X)}$ is deduced from the uncertainty of distance X, u_X, and of distance X_0, u_{X0}, by the equation:

$$\frac{u_{F_{(x)}}}{F_{(x)}} = \left[2*\left(\frac{u_x}{x}\right)^2 + 2*\left(\frac{u_{x_0}}{x_0}\right)^2 \right]^{1/2}$$

Applying this equation to the worst case, i.e. efficiency measured at 150 cm and a sample measured at 50 cm, and taking an uncertainty of distance to be 1 cm, we get:

$$\frac{u_{F_{(x)}}}{F_{(x)}} = \left[2*\left(\frac{1}{50}\right)^2 + 2*\left(\frac{1}{150}\right)^2 \right]^{1/2} = 3\%$$

Attenuation correction

The calculation code allows the user to take into account the attenuation of the various known screens (physical and chemical make-up, density and thickness) that might be placed in front of the sample during measurement, but not when measuring the efficiency. In practice, this situation is fairly rare when characterising uranium waste drums, because screens that decrease the intensity of the photon emission rate are not used, given that the rate is usually low enough not to saturate the electronic detection and counting systems.

However, for some special cases where screens of known thickness and composition are used, the uncertainty associated with attenuation correction remains low. By experience, the value used for this uncertainty, which is an overall value, is u_{att}/C_{att} **= 2**%.

Self-absorption correction

Self-absorption corrections that are not taken into account by extrapolation to infinite energy and that are due to materials other than those with high atomic numbers (uranium, plutonium, thorium,…), can reach 50% for some waste drums. Since the factor µX, which is the attenuation of the photon emission rate passing right through the drum, is not known beforehand, a measuring device (scanning γ) needs to be used to measure it.

The experimental uncertainty associated with the use of the scanning γ device to determine the correction is of the order of 4%. So we use u_{aut}/C_{aut} **= 4 %**.

4.3. Uncertainties determined by a type A method

Repeatability

Repeatability was studied under the following conditions

- One source from the 20 reference samples used by IRSN [5]
- Without matrix
- Measurement distance: 20 cm
- Mass of uranium: 12.46 g
- Enrichment in ^{235}U: 5.49%
- Measurement time: 150, 300, 600, 1200 and 3600 s
- Number of repetitions for each measurement time: 50

For a net count of approximately 320 counts at 1001 keV, the relative standard deviation of the 50 measurements is close to 8%, and can be considered as an estimate of the relative uncertainty of repeatability.

This uncertainty includes the count rate uncertainty u_{compt} / ($N_{(E)}$ – $B_{(E)}$). So the count rate uncertainty should not be counted when calculating the overall uncertainty.

Photon emission rate per mass unit

For the isotope in question, ^{235}U or $^{238}U/^{234m}Pa$, the photon emission rate per unit mass $P_{\gamma(E)}$ is calculated from the probability of gamma emission $p_{(E)}$ with energy E, by the equation:

$$P_{\gamma(E)} = \lambda \times N \times p_{(E)} = \frac{Ln2}{T} * \frac{6.02 \cdot 10^{23}}{m} * p_{(E)}$$

m and T are the atomic mass and the half-life respectively of the isotope in question.

The uncertainty of the photon emission rate per unit mass $u_{P\gamma}$ is included in the uncertainty of accuracy, the calculation of which is explained below. So this uncertainty should not be counted when calculating the overall uncertainty.

Accuracy

The self-absorption corrections due to uranium $C_{U(E)}$, and more generally to high atomic number materials, can be a correction factor of over two for certain waste drums. They are counted in the extrapolation to infinite energy method.

On the other hand, linear extrapolation to infinite energy leads to an overestimation which is a function of the surface density of the uranium sample measured. This phenomenon is due to the fact that the equation relating self-absorption to the inverse of energy is not strictly linear, and has different curves depending on the surface density of the sample. Conversely, above a surface density of 3 g/cm^2, infinite thickness is reached at energies of approximately 300 keV and then the mass of nuclear matter is underestimated.

It is no easier to evaluate the underestimation as a function of surface density than it is to fit a model that is more accurate than the linear model as a function of surface density. In the first case it would take a huge number of experiments, and in the second case the small number of pairs of points available means that no other regression types can be envisaged.

Therefore the uncertainty associated with the method of extrapolation to infinite energy needs to be integrated into the systematic error of the method. In the present case, this was done experimentally by a study of the accuracy of the method. This experimental study shows that the measurement method tends to underestimate by an average of 25%, with a maximum discrepancy of 50% observed on one of the reference samples.

Considering that the reference samples used lead to an observable maximum overestimation, and considering a normal distribution law for the overestimation, the uncertainty associated with self-absorption of uranium $C_{U(E)}$ can be estimated at approximately 50%/3, i.e. approximately **17**%.

A plausible explanation lies in the poor fit of the linear regression model with the real situation. The self-absorption phenomena represented using the logarithm of the apparent mass of uranium as a function of the inverse of energy produce various curves as functions of the surface density of the sample, and on extrapolation to infinite energy, lead to a variable overestimation of the mass. Given its size, since the calculation of the mass of uranium cannot be corrected, the uncertainty of accuracy needs to be introduced into the calculation of the mass of uranium.

Isotopic composition

One particular assay was selected during which the fluctuations in calculating the isotopic composition were deemed to represent the mean fluctuation that can be observed during acquisitions on waste drums.

During this assay, the standard uncertainty observed in the distribution of values of enrichment in uranium 235 was approximately 8.2%. Applying the MaPU method taking account of each of the enrichment values determined previously produces a family of values of the mass of uranium. The standard uncertainty observed for the distribution of masses of material determined by applying the MaPU method is approximately 2.5%.

This value of 2.5% is deemed to be representative of the standard uncertainty caused by the usual dispersion expected for $K_{(E)}$ in the mass of uranium calculated by the MaPU method.

Overall relative uncertainty

The principle of error propagation is applied, assuming that:

- No significant correlation exists between the different influencing factors.
- The systematic error is estimated by the study of the accuracy of the method.
- The uncertainty of repeatability of $\frac{u_{rep}}{rép}$ includes the uncertainty of count rates $\frac{u_{(N_{(E)}-B_{(E)})}}{(N_{(E)}-B_{(E)})}$.
- The uncertainty of accuracy of $\frac{u_{just}}{just}$ includes the uncertainty of photon emission rate per mass unit $\frac{u_{P_{\gamma(E)}}}{P_{\gamma(E)}}$ and the uncertainty of extrapolation to infinite energy $\frac{u_{C_{u(E)}}}{C_{U(E)}}$.

The overall relative uncertainty of the mass of uranium then becomes:

$$\frac{u_M}{M}=\left[\left(\frac{u_{\varepsilon(E)}}{\varepsilon(E)}\right)^2+\left(\frac{u_{K(E)}}{K(E)}\right)^2+\left(\frac{u_{C_{att(E)}}}{C_{att(E)}}\right)^2+\left(\frac{u_{C_{aut(E)}}}{C_{aut(E)}}\right)^2+\left(\frac{u_{C_{(geo)}}}{C_{(geo)}}\right)^2+\left(\frac{u_{F_{(x)}}}{F_{(x)}}\right)^2+\left(\frac{u_{C_{pile-up}}}{C_{pile-up}}\right)^2+\left(\frac{u_{rép}}{rép}\right)^2+\left(\frac{u_{just}}{just}\right)^2\right]^{1/2}$$

which is

$$\frac{u_M}{M}=\left[3^2+2{,}5^2+2^2+4^2+3{,}4^2+3^2+2^2+7^2+16{,}7^2\right]^{1/2}\approx 20\%$$

The following table summarises the uncertainties of measuring the mass of uranium in waste drums:

Influencing factors or characteristics of the method	Relative uncertainty (%)
Counting: uncertainty defined from the uncertainty of repeatability	-
ε: total absorption efficiency of the gamma spectrometry chain	3
K: Isotopic ratio	2.5
P_γ: photon emission rate per mass unit, defined from the uncertainty of accuracy	0
C_{att}: screen attenuation correction	2
C_{aut}: self-absorption correction due to materials other than uranium	4
C_{geo}: correction for geometry	3.4
F: efficiency response curve transfer correction	3
$C_{pile\text{-}up}$: correction of count loss due to pile-up	2
C_U: correction for self-absorption due to uranium, defined from the uncertainty of accuracy	-
Repeatability	8
Accuracy	16.7
Overall for the mass of uranium	**20**

A correlation exists between the relative uncertainty of counting and the relative uncertainty of repeatability of the mass of uranium. Although it is possible to increase the count time and thus decrease the uncertainty of repeatability, this would only slightly decrease the overall relative uncertainty of the mass of uranium, since the preponderant factor of the overall relative uncertainty is the uncertainty of accuracy. This justifies the fact of opting for counting 320 counts at 1001 keV to estimate the uncertainty of repeatability and the acquisition time of a spectrum.

4.5. Validating the uncertainty of measurement

This validation consists in applying a test comparing the measured result of each reference sample with its reference value [7]. The test is described as follows, assuming that the uncertainty of reference samples is negligible compared with the uncertainty of the measured result:

$$-u \leq \frac{\Delta}{\sigma} \leq +u$$

Δ is equal to the measured result minus the reference value, and u is chosen to be 3, which leads to an alpha risk of 0.26%, which is the probability of declaring a non-existent discrepancy.

This test is applied to all the reference sample measurements. The results obtained show that the maximum value of $\frac{\Delta}{3\sigma}$ is 0.6 and it never exceeds 1, which would have led to the conclusion that the measured result is different from the reference value, with a risk of error of 0.26%. This validation shows that the overall uncertainty estimation of 20% of the measured result is satisfactory.

5. Measurement time and stopping criteria

One way to evaluate the performance of a measurement method is to evaluate, via experiments, certain features such as:

- Selectivity, specificity
- Measurement accuracy
- Repeatability
- Reproducibility
- Linearity
- Robustness
- Scope of applicability.

IRSN has done experiments to determine accuracy, repeatability and scope of applicability. Since there is a correlation between some of these parameters and the spectra acquisition time used to calculate the mass of uranium, a special study was made of acquisition time.

Since measurement time has only a small influence on overall relative uncertainty of measurement, a measurement time needs to be chosen such that the uncertainty of repeatability is acceptable. The experimental study showed that the spectrum acquisition time needed to obtain 320 counts at 1001 keV is 150 s for a drum - detector distance of 20 cm.

As a first approximation, this gives an acquisition time of $\left(\left(\frac{150}{20}\right)^2 * 150\right) / 3600 \approx 2.5\,\text{hours}$ for a drum - detector distance of 150 cm.

Considering the time needed to set up the waste drums, 3 or 4 measurements per day is a reasonable estimation.

One practical recommendation is to evaluate, at the start of acquisition, the time needed to obtain at least 320 counts in the 1001 keV peak.

The criterion for stopping spectrum acquisition needs to meet two vital conditions:

- The number of counts in the net surface area of the $^{238}U/^{234m}Pa$ peak at 1001 keV must be at least 320.
- The measured results as a function of the acquisition time must be stable, which will be achieved using the acquisition and spectrum analysis supplied by the AutoISO_PLUM code. One way this stability is achieved is when the measurement of uranium 235 enrichment itself is stable.

6. Scope of applicability - Conclusion

The scope of applicability of the MaPU code applied to measuring the mass of uranium in waste drums is as follows:

- ^{238}U and ^{234m}Pa have reached secular equilibrium
- Absence of ^{226}Ra to avoid interference with the ^{226}Ra peak at 186.21 keV and the ^{235}U peak at 185.7 keV.
- Any physical or chemical form.
- Any enrichment, but preferably weak enrichment, because the ^{238}U peak at 1001 keV needs to be available for use.
- Surface density of the sample not exceeding $3 g/cm^2$.
- Any matrix as long as the photons from ^{152}Eu at 121 keV used for scanning are not completely attenuated (to allow for the waste drum µX measurements).
- It is advisable for both 766 and 1001 keV peaks to be present, notably for the linear regression. One of them is mandatory.
- At least 320 counts in the $^{238}U/^{234m}Pa$ 1001 keV peak.
- Negligible background noise compared with the sample. If not, it must be reduced. Use of a protection around the detector is recommended so that the MeV energy photons are negligible compared with the sample photons.

- Measurement distance at least approximately 150 cm.

Within this scope of applicability, MaPU can measure the mass of uranium packed in waste drums with a relative uncertainty of measurement of 20%.

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SESSION 18

PROCESS MODELLING AND DATA TREATMENT

International Target Values for Measurement Uncertainties in Safeguarding Nuclear Materials: preparing for the 2010 Update

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Abstract:

"International Target Values (ITVs) for Measurement Uncertainties in Safeguarding Nuclear Materials" were first published by the IAEA in 1993. A Consultants Group had recommended to implement the concept of the "ESARDA Target Values", as introduced earlier by the ESARDA Working Group on Techniques and Standards for Destructive Analysis. The ITVs bear a date and are expected to be revised to reflect the experience that measurement quality may improve with the development of newer methods and instruments. The first revision of the 1993 issue was published in April 2001 as the "ITVs 2000". The IAEA now started to prepare for the next update in 2010.

Most likely, the largest data sets of actual measurement results from industrial types of materials are contained in the IAEA's database consisting of operators' declared and inspectors' verified data for various types of materials and measurement techniques. The present paper gives examples of actually observed measurement quality for DA and NDA based on the statistical evaluation of operator-inspector differences from more than 20 years. The derived uncertainty estimates will be a major source of information for the revised Target Values.

Keywords: target values, measurement uncertainties, safeguards, nuclear material

1. Introduction

Safeguarding nuclear material includes a quantitative verification of the accountancy of fissile materials by independent measurements. The effectiveness of these verifications depends to a great extent upon the quality of the accountancy measurements achieved by both the facility operator and the safeguards inspectorate. For this reason a typical model Safeguards Agreement[1, 2] stipulates that:

> *"The Agreement should provide that the system of measurements on which the records used for the preparation of reports are based shall either conform to the latest international standards or be equivalent in quality to such standards".*

Although the above requirement was directed to the facility operators, it indeed applies equally well to the safeguards inspectorates.

In the absence of relevant international standards of measurements, the International Atomic Energy Agency (IAEA) had defined in the 1970s a set of international standards of nuclear material accountancy[3], which lists the "expected measurement accuracy associated with the closing of a material balance" at five different types of nuclear facilities. However, these values have never been reviewed despite numerous technological changes since their adoption by consensus by a group of experts designated by their Governments. Safeguards officials and evaluators, but also plant measurement specialists, need more current and informative references regarding the performance capabilities of measurement methods used for the determination of the volume or mass of a material, for its sampling, and for its elemental and isotopic assays. It was therefore that the IAEA adopted the concept of International Target Values (ITVs) to define the measurement quality that should be achievable under the conditions normally encountered in typical industrial laboratories or during actual

safeguards inspections. The ITVs bear a date in order to reflect the experience that the quality of measurements may improve with the development of newer methods and instruments and should therefore be updated at regular intervals.

Most likely, the largest data sets of actual measurement results from industrial types of materials are contained in the IAEA's database consisting of operators' declared and inspectors' verified data for various types of materials and measurement techniques. This paper describes the history of the "Target Values" and gives examples of actually observed measurement quality for destructive analysis (DA) and non-destructive assay (NDA) methods based on the statistical evaluation of operator-inspector differences from more than 20 years.

2. The History of the "Target Values"

The Working Group on Techniques and Standards for Destructive Analysis (WGDA) of the European Safeguards Research and Development Organisation (ESARDA) presented in 1979 a list of "Target Values" for the uncertainty components in destructive analytical methods[4] to the Safeguards authorities of Euratom and of the IAEA. After four years of extensive discussions revised estimates were prepared in collaboration with operators' laboratories and safeguards organizations and published as the 1983 Target Values[5]. The international acceptance grew further with the next review which involved the active participation of the members of two specialized committees of the Institute of Nuclear Material Management (INMM). The 1987 Target Values[6], published as result of this review, defined, like the previous editions, the values of "random" and "systematic" error parameters to be aimed for in element and isotopic analysis of the most significant types of materials using common destructive analytical methods. The same group expanded the scope of the Target Values with the 1988 edition[7] in defining values of the random error parameter to be met in the elemental assays as a result of sampling.

Following a 1988 recommendation of the IAEA Standing Advisory Group on Safeguards Implementation (SAGSI), the IAEA convened a Consultants Group Meeting in June 1991 to provide expert advice on international standards of measurements applicable to safeguards data. A concept of ITVs was proposed on the model of the 1988 ESARDA Target Values and included estimates of the "random and systematic error" uncertainties originating from the measurements of volumes or masses of nuclear materials. The scope of ITVs was also extended to include a consideration of the NDA methods which had won acceptance as accountancy verification tools.

Specialists from four continents took part in the discussion of the proposed concept. The result was the publication of an IAEA Safeguards Technical Report in March 1993, titled "1993 International Target Values for Uncertainty Components in Fissile Isotope and Element Accountancy for the Effective Safeguarding of Nuclear Materials"[8]. Articles in the ESARDA Bulletin[9] and in the Journal of the INMM[10] widely publicized the IAEA technical report. The report itself was translated into Japanese[11].

In 1999 and 2000 international experts and panels then reviewed the experience gained with the use of the 1993 ITVs and the progress made since 1993 in accountancy and safeguards verification measurements. Furthermore an effort was made to bring the nomenclature in line with the recommendations of ISO[12], the National Institute of Standards and Technology (NIST)[13] and the European Association of Chemical Measurements (EURACHEM)[14]. A clear distinction for example was made between the meaning of the term "error" and the term "uncertainty".

The ITVs 2000 were published as an IAEA Safeguards Technical Report in April 2001, titled "International Target Values 2000 for Measurement Uncertainties in Safeguarding Nuclear Materials"[15]. Again, the ITVs 2000 were widely publicized through articles in the ESARDA Bulletin[16] and in the INMM Journal[17].

The ITVs bear a date in order to reflect the experience that the quality of measurements may improve with the development of newer methods and instruments. The Target Values should be achievable under the conditions normally encountered in typical industrial laboratories or during actual safeguards inspections. They do not represent the measurement uncertainties, which would only be achieved under exceptional or ideal laboratory conditions, or with most recently developed methods, which have not yet found wide use for daily and routine measurements. As such, Target Values should be derived from an evaluation of actual measurement data.

The present paper gives examples of actually observed measurement quality for DA and NDA based on the statistical evaluation of operator-inspector differences from more than 20 years. The derived uncertainty estimates will be a major source of information for the preparation of ITVs 2010.

3. The IAEA's Operator-Inspector Database

The IAEA's database contains a large volume of verification data related to both DA and NDA. Historical data are available for verifications by DA since 1981 and for verifications by NDA since 1988.

For a particular item selected for verification by DA, which involves the taking of a representative sample and its shipment to a laboratory, the Operator's declared data will generally be for the mass or volume of material, the U and/or Pu element concentration(s), the ^{235}U abundance and/or Pu-isotopic composition. The Inspector's verification data will be the mass or volume of material, measured at the facility, and the corresponding results obtained by the analytical laboratory for sample received.

For an item selected for verification by NDA, the Operator's declaration will be mass of material, U elemental mass and ^{235}U in case of U-materials, or mass of material and Pu elemental mass in the case of Pu-materials. The corresponding Inspector's data will then be the mass of material and, depending on the material type, the ^{235}U abundance or the mass of Pu element.

The "raw data", as declared by the Operator, reported by the analytical laboratory or measured by the Inspector, are then converted into common units and, in the case of Pu materials, decay corrected to the inspection date. Results of replicate measurements are averaged.

These Operator-Inspector paired data are labelled with the corresponding Material Balance Area (MBA) and Stratum (material type) codes and are thus accessible for paired comparison and material balance evaluations.

4. Verification Measurement Performance Evaluations

The evaluation system is based on Operator-Inspector paired data to yield estimates of random and systematic error standard deviations. A basic assumption is that the random and the systematic uncertainty components are characteristics of the type of material, its chemical and physical form and of the method of measurement. A further assumption is that the component of systematic character is constant for a given period, but that it varies in a random manner from one inspection to another, for both the Operator and the Inspector.

In the IAEA data analysis, various statistical techniques[18] are used to derive separate estimates of the Operator's and Inspector's uncertainty parameters based on the collection of historical Operator-Inspector differences. The results of these evaluations are "Performance Values" obtained for each MBA/stratum/measurement method combination.

These estimates reflect the actually observed "verification measurement performance" and include all sources of errors (e.g., sampling, analysis, and also transcription errors and potential falsifications which are small enough to escape identification in the outlier test).

For the examples of Verification Measurement Quality given below, the individual performance values from all MBAs for a given material type and measurement method were pooled. The values given as random and systematic uncertainties are the medians of the population of individual estimates.

Figure 1, on the example of ^{235}U measurements on low enriched UF_6 with Ge detector based NDA instruments illustrates this.

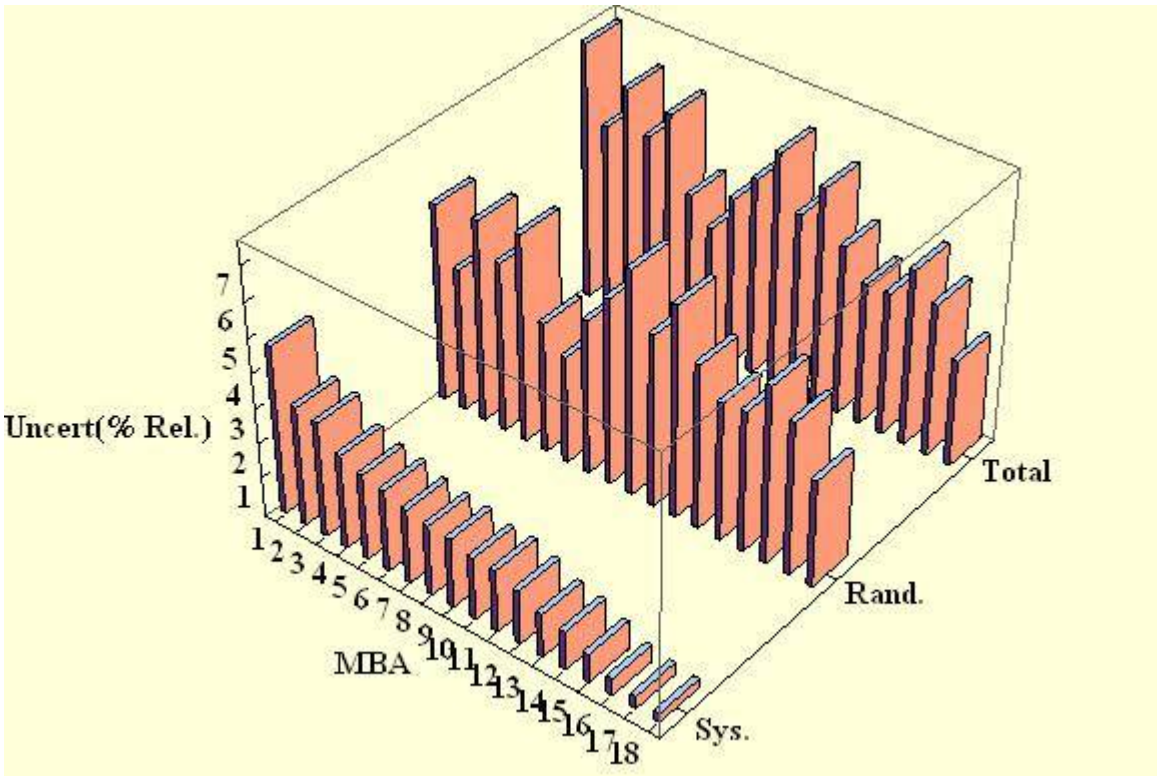

Figure 1: ^{235}U measurements on low enriched UF_6 by NDA (Ge detector based instruments)

5. Examples of NDA Measurement Quality

Verification measurements by NDA represent a special situation in that Operator's declared data, based on DA, are compared with much less precise and/or accurate Inspector's results based on NDA. It is assumed that the total fluctuation originates practically solely from the Inspector's measurements and therefore separate estimates of the Operator's and Inspector's uncertainty parameters cannot be derived.

Table 1 summarizes the uncertainty estimates for NDA measurements on UF_6 cylinders, containing low enriched (LEU), natural (NU) and depleted (DU) material. The estimates further distinguish between Ge-detector (*MCG) and NaI-detector (*MCN) based NDA systems.

Material Type	Instr.	MBAs/ Strata	n	Random (Median, %rel.)	Systematic (Median, %rel.)
LEU	*MCG	18	11290	4.8	1.8
	*MCN	10	1327	4.8	2.2
NU	*MCG	7	1406	12	1.3
	*MCN	7	3684	13	2.2
DU	*MCG	7	478	19	8.1
	*MCN	8	2960	22	3.1

Table 1: ^{235}U-abundance in UF_6 by NDA

Similar evaluations have been made for

- ^{235}U abundance measurements in other uranium materials (LEU oxides, pellets, rods and scraps, HEU and NU oxides), applying Ge- and NaI-detector based instruments;
- ^{235}U-total measurements in HEU compounds (by Active Well Coincidence Counter - AWCC), in LEU rods (by Fuel Rod Scanners – FRSC), and in LEU fuel elements (by Uranium Neutron Coincidence Collar – UNCL).

Table 2 gives uncertainty estimates for various types of Pu materials, using the neutron measurement based instruments HLNC (High-level Neutron Coincidence Counters), PSMC (Plutonium Scrap Multiplicity Counter) and INVS (Inventory Sample Counter for measurement of small samples).

Instr.	Material Type	MBAs/ Strata	n	Random (Median, %rel.)	Systematic (Median, %rel.)
HLNC	Pu-oxide	10	14096	1.5	0.40
	FBR-MOX	4	3814	2.3	0.79
	LWR-MOX	6	3222	5.5	1.5
	LWR Rods	5	2030	2.2	0.59
	MOX Scrap	8	1609	5.8	1.0
PSMC	Clean MOX	7	619	3.8	1.2
	Scrap	6	1162	6.2	0.89
	Scrap	4	295	3.4	0.28
INVS	All MOX	28	4935	4.8	1.8

Table 2: Pu-total in Pu materials (n-measurement based instruments)

Similar evaluations have been made for facility specific instruments to determine Pu-total in glove boxes (hold-up measurements), in waste drums, in MOX canisters, FBR-MOX fuel pins and FBR-MOX fuel assemblies.

6. Examples of DA Verification Measurement Quality

For verification measurements by DA it is possible to derive separate estimates of the Operator's and Inspector's uncertainty parameters based on the collection of historical Operator-Inspector differences. The separation of the error between Operator and Inspector is made possible by assuming that the verified items are selected randomly. Separate estimates are of particular interest in the interpretation of the observed measurement performance, because in many instances the Operator declared values for element concentrations and ^{235}U abundance are based on measurement techniques different from those applied for the analysis of the Inspector samples.

Furthermore it should be noted that the Operator and Inspector generally analyze on independent samples and therefore "sampling uncertainties" are contained in the measurement performance

estimates. For instance, sampling errors for fresh UO_2 powders should be larger than those to be expected for sintered pellets.

Table 3 summarizes the uncertainty estimates for U-concentration measurements in the most commonly encountered uranium materials. The analytical technique applied for the Inspector samples is generally potentiometric titration, while the Operators generally apply ignition gravimetry in combination with impurity analysis.

Material Type	MBAs/ Strata	n	Random (Median %rel)		Systematic (Median %rel)	
			Operator	Inspector	Operator	Inspector
Powders	30	3201	0.12	0.13	0.036	0.037
Pellets	26	2708	0.024	0.035	0.011	0.0173
Pellets (Gd)	12	453	0.058	0.081	0.013	0.023
Scrap	23	1062	0.086	0.080	0.032	0.032

Table 3: U concentration in U materials by DA

Table 4 lists the uncertainty estimates for ^{235}U enrichment measurements in the major LEU materials. The analytical technique applied for the Inspector samples is generally thermal ionization mass spectrometry. Some of the Operators in LEU fuel fabrication facilities apply gamma spectrometry, while in enrichment facilities gas source mass spectrometry is used. The expected measurement performance of these techniques differs.

Material Type	MBAs/ Strata	n	Random (Median %rel)		Systematic (Median %rel)	
			Operator	Inspector	Operator	Inspector
Powders	19	2104	0.20	0.25	0.093	0.073
Pellets	23	2286	0.22	0.20	0.094	0.073
UF_6	8	959	0.10	0.092	0.048	0.042
Scraps	9	465	1.4	0.95	0.20	0.17

Table 4: ^{235}U enrichment in LEU materials by DA

Table 5 summarizes the uncertainty estimates for Pu-concentration measurements in the most commonly encountered plutonium materials. The analytical techniques applied by the Operators and for the Inspector samples cannot clearly be differentiated. Since several years Isotope Dilution Mass Spectrometry has won wide acceptance, while earlier potentiometric titration and coulometry were used.

Material Type	MBAs/ Strata	n	Random (Median %)		Systematic (Median %)	
			Operator	Inspector	Operator	Inspector
Pu-nitrate	5	1248	0.34	0.26	0.22	0.23
Pu-oxide	2	305	0.06	0.29	0.12	0.20
FBR-MOX	10	751	0.70	0.75	0.18	0.31
LWR-MOX	4	302	0.65	0.70	0.48	0.63
MOX-scrap	3	82	1.4	1.4	0.31	0.46

Table 5: Pu concentration in Pu materials by DA

7. Conclusion

The IAEA's Operator-Inspector data base represents most likely the largest data set of actual measurement results from industrial types of nuclear materials. The results of the verification measurement performance evaluations for DA and NDA measurements, described and presented in this paper, are based on the statistical evaluation of operator-inspector differences for more than 20 years. The derived uncertainty estimates will be a major source of information in the process of revising the existing ITVs 2000 in preparation for the ITVs 2010.

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Design, Diagnose, Deliver – An NRTMA Approach to Plant Measurement Performance

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Abstract:

The performance of any Near Real Time Materials Accountancy (NRTMA) system depends on the theoretical correctness of the analytic approach and the quality of the data that is provided by the plant. The first of these has been dealt with extensively elsewhere. This paper will look at how the Sellafield Ltd. NRTMA[1] system based on Page's Test[2] can be used to determine the effectiveness of interim assurance of a new plant based on the plant performance parameters proposed in the design documents.

Keywords: NRTMA, Measurement, Performance, Control.

1. "NRTMA should tell you nothing!"

This was the often heard comment from the NRTMA system design team. It was not a directive to be parsimonious with information but a statement of ultimate NRTMA truth.

For a plant to be demonstrably in control in Near Real Time it must meet four conditions;

- The plant must maintain appropriate measurement systems for all material flows and locations within the compass of the NRTMA system;
- The plant must have appropriate control systems to collect and analyse this data;
- The business rules and mathematical modelling, upon which the NRTMA system is based, must reflect the operation of the plant, and
- The plant must be run in such a manner that the data analysis system (NRTMA) shows no alarms.

It was to the above aspirational plant state, total measurement control, that the quotation of the section title referred. If everything were as the designers of the plant and the NRTMA system, would have it be, then NRTMA will 'tell you nothing' because there is nothing to tell! The function of NRTMA is to alarm when things are not right due to, possibly, inappropriate mathematical models, inappropriate measurement uncertainties or even an apparent or, heaven forefend, actual diversion.

Herein lies the problem addressed by this paper. Measurement matters. The task is to provide sufficient measurement control without letting costs spiral out of control.

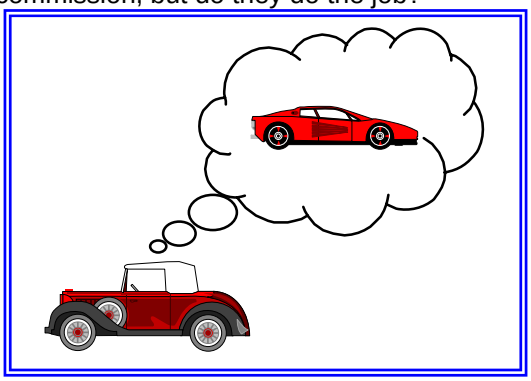
When building a measurement control system, a designer is going to ask the materials accountant the question, "How good do you want the measurements to be?" A typical answer that then comes back is "The best you possibly can…" This is not only unresponsive, it is very unhelpful.

At this point an engineering design department working on weighing is entitled to send back a set of plans that contain a measurement system consisting of a pair of

[1] Any further reference to NRTMA within the context of this paper should be taken as meaning a reference to the Sellafield Ltd. NRTMA system based on Page's Test

[2] This paper is not intended to address the issue of 'What is Page's Test?' or be a description of the Sellafield Ltd. NRTMA system. These have been covered extensively in the literature already. The reader is referred to the references 1→5. There is also a good summary of the Sellafield Ltd. approach to NRTMA in the appendices of reference 4 and a description of the two original NRTMA systems in reference 5.

bathroom scales.[3] As a solution for the engineers, they are easy to procure and commission, but do they do the job?



There is an obverse to this philosophical coin. The designers can (and do!) approach project sponsors with the offer that they can turn the supposedly model-T measurement system into a formula-1, measurement system for a very modest sum. The term modest here is being used with its alternative meaning of – it is going to cost a fortune; the project time will slip and cost over-run is a certainty.

Caught between the rock of designer ambition and the hard place of spiralling measurement control cost such a project manager faces difficult choices. It seems that they must accept the potential for a system that may not meet accountancy requirements or one whose cost is not justified by gains in performance. Neither is a terribly comforting outcome. Fortunately, Page's Test can be used to provide just such a judgement aid and enable a balancing of performance versus cost.

2. Measurement Control - A short review

What goes into a process must come out! One might think "How hard can this be?" When counting cans or cars or anything that comes in discrete packages, this shouldn't represent too much of a challenge[4]!

A space in a full car park should not be difficult to observe though it may be difficult to find. The real challenge comes when items are being processed

In a process area the measured Inputs and Outputs provide a book balance (BB[5]). This is what you think you have. The potential for crisis comes when you compare this with what you can actually find (PIT[6]). The Inventory difference (ID[7]) then has to be tested for significance. Is this a "real" inventory difference or one subject to measurement uncertainty or possibly an accountancy mistake?

3. Performance against Cost

Having weighed herself one month ago at 60kg, Miranda, finds that in the following month she weighs half a kilogram less. Delight all round! A further month on, sadly, she finds that she has 'put on' half a kilogram and returned to her previous weight.

The reality may well be that nothing actually changed at all. Her happiness (and sadness)

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[3] The authors are merely alluding to the well-known fact that no bathroom scales ever provide the information that the user expects to see. Scales are always biased on the heavy side.

[4] In an original draft it suggested that counting 'doesn't represent too much of a challenge.' One of the authors then spent some time watching the 'less than ten items' queue in a local supermarket. It is clear that, for a lot of people, counting can be seriously challenging.

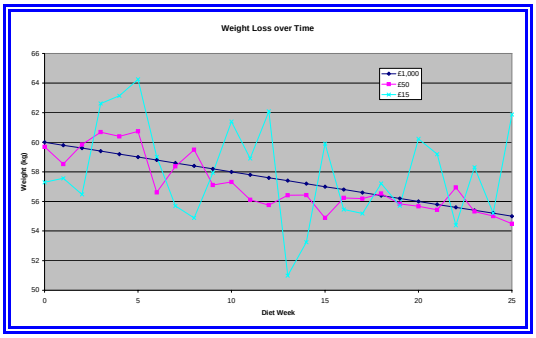
[5] In nuclear materials terms the book balance is defined as BB = Opening Inventory + Receipts - Issues

[6] **P**hysical **I**nventory **T**ake. This is the nuclear industry equivalent of a very detailed stock check usually carried out annually at the end of a campaign of production run. Where these stock estimates are run more frequently without a plant shut down, they are called **I**n-**P**rocess **I**nventories (IPIs).

[7] The **I**nventory **D**ifference (ID) given by PIT-BB will be negative if there is a loss and positive if there is a gain.

was simply a function of the ability of her bathroom scales to measure with a suitable accuracy and precision. Six months later, after joining a health club, her scales inform her that she weighs 55 kilos. Pure joy! But is her confidence that the difference is real, misplaced?

Measurement control is therefore a function of quality of measurement as well as the ability to take measurements. There is also another aspect to consider, in this case, the cost of the scales. Suppose Miranda, above, whilst dieting, monitors her weight frequently. The results of using various costs of scales are shown on the graph.



The £1000 model (dark blue, accuracy better than ±1g), is clearly overkill, however the £15 model (light blue, accuracy ±3kg) doesn't seem very satisfactory.

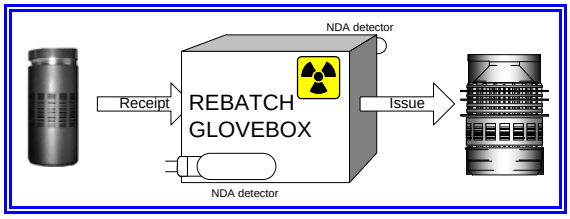
Looking at the graphs, it is clear that if Miranda were to purchase the high precision and high cost scales she would know very soon after that her diet strategy was indeed working. However, if she were to purchase the very cheap and inaccurate scales it would be very difficult for her to establish whether her diet was working or it was in fact the poor quality of her scales that was to blame for the apparent lack of weigh loss. The issue here is what quality of scales does she need to purchase to give her the performance that she wants (i.e. measures significant weight loss) and unfortunately she will only know this once she has purchased the scales i.e. after the event!

4. Pages Test

Miranda has a day job. She is part of a team designing a nuclear fabrication facility. Her particular brief involves the material control of a rebatching system and more specifically the specification for the performance requirements of the Non-Destructive Assay (NDA) detection system.

The approach to determining an appropriate NDA performance specification can be divided into a number of steps but the overall intent is to provide the information required to calculate a Page's Test and then evaluate the NRTMA system response of this test to a variety of loss/gain scenarios. To do this, Miranda has to obtain design time values for the following:

- Opening & Closing Inventory Standard Deviation
- Throughput Standard Deviation Standard Deviation
- In Process Inventory (IPI) Standard Deviation.
- Campaign Length
- Timeliness
- False Alarm Probability (FAP) (Red, Amber & Weighting (Protracted/Abrupt ratio))



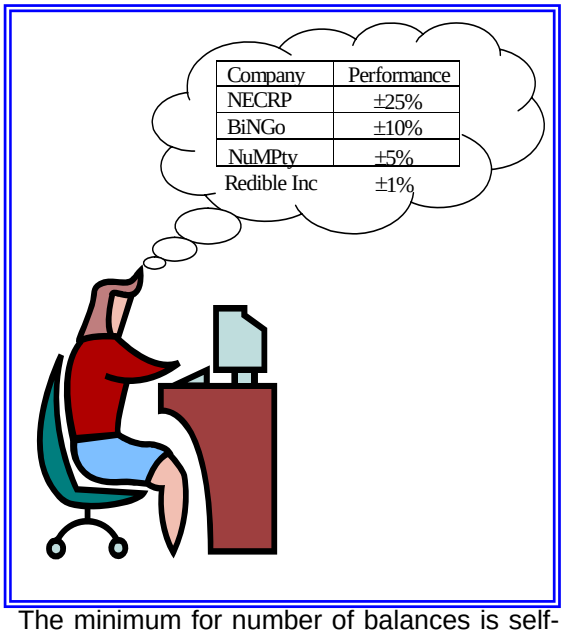
The estimates for the standard deviations for Throughput[8], Opening, Closing and In Process Inventories can be made based on plant design parameters and, in the case of Opening and Closing Inventories, could be set at zero by assuming a full plant clean out. Campaign length is chosen to be one year in view of the necessity of holding a PIT every year and timeliness is also determined by regulation[9]. FAP and weighting are a decision for the Company to make, depending on the performance requirements of the detection system. Miranda is left with

[8] The values chosen here and throughout have no particular significance and were chosen for ease of use in the illustrations. The calculation is in Appendix 2.

[9] The campaign length is a function of operational requirements and may be less than a year but regulation ensures that it cannot be greater.

only one final detail to estimate; the In-Process Inventory (IPI) standard deviation.

Given the relative errors for the NDA equipment this means some research in the design documents for an average design IPI, which, for the sake of this example, is 20kg. There only remains a decision on the precision requirements for the NDA detection and the number of material balances to be calculated during the campaign[10]. Looking at the general market place gives a range of values for the performance of the NDA detection system.



The minimum for number of balances is self-evident. There has to be at least one balance per timeliness period if detection of a goal quantity is to be possible. A value for the maximum number of balances per timeliness period is less obvious. Clearly there will be an upper limit to this number of balances which could be related to such factors as; recycle times on the NDA equipment, local system controller usage and availability, the ability of a computer system to process the data[11].

5. Res Miranda

Miranda now has sufficient information for an assessment to be made of the performance of the NRTMA system for the various NDA performance scenarios e.g. IPI estimate standard deviations of 25%, 10%, 5% & 1%. A full list of the required tests is given in Appendix 1.

The evaluation shows the response of the NRTMA system, using the specified Page's Test. This response is in two parts:

- The effect of an abrupt loss of material in a single timeliness period and
- The average loss of material in a campaign for a protracted loss based on a goal quantity of material spread throughout the campaign[12].

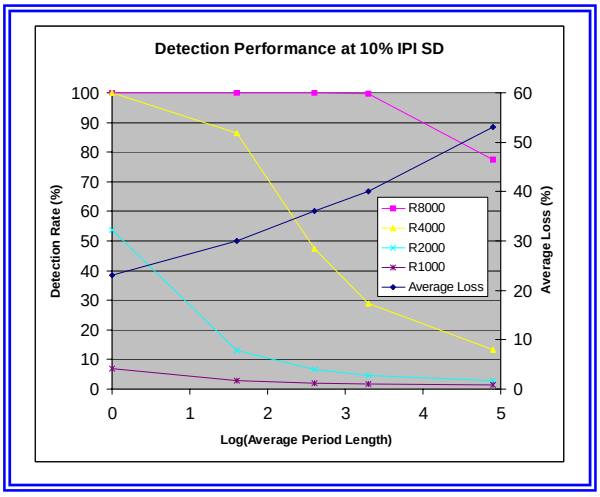
Once the results for the evaluations are compiled , see appendix 2, it is possibly surprising to note that even a 25% random error in IPI measurement will yield a near perfect score for diversion detection at a frequency of 'once per day'. Nonetheless this is clearly an unsatisfactory choice as it would leave no room for variation in assumptions.

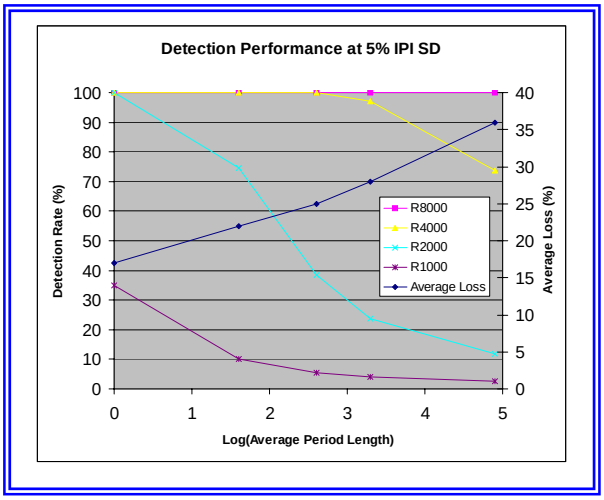
Similarly, after perusing the journals, Miranda may conclude that 1% represents a designer's aspiration rather than an achievable target with current technology and any cost would probably be inflated by the need to cover additional development.

[10] The total number of balances is a function of campaign length, timeliness – that is the period within which detection is expected to occur – and the number of balances taken in that detection period. It is assumed that campaign length and timeliness are accessible givens before plant design begins.

[11] Current estimates for calculations over <n> periods suggest that the time is worse than $\Theta(n^2)$. Over a whole campaign 300 periods certainly has an acceptable calculation time but 3000 periods probably does not. However, as this paper will show, this may not be an issue.

[12] See reference 2





However, what is clear is that anything approaching 5% is going to have significant capabilities above and beyond detecting goal quantities. This then allows Miranda, having determined an optimum range for performance (5→10%), to investigate other assumptions in a series of what if…? scenarios, e.g.

What if …

- … the can weight measurement were better/worse?
- … the analytical analysis performance were better/worse?

Additionally, it is clear from the results that a frequency for IPIs of greater than one per day does not provide any particular value added. Surely 100% detection probability is good enough?

6. A Small Digression

Clearly ten IPIs per day sounds wonderful, if possible and assuming that the software can handle this much data. However, this does not, as stated above, give any value added over once per day. The objective of NRTMA is the timely detection of diversion of a goal quantity (GQ) of Special Nuclear Material. In this respect timeliness period is set at thirty days. Pragmatically, the software used for this paper can set this to a fortnight or a week or even a day in preference to the stated thirty days. Similarly the goal quantity can be set to smaller values. In this latter case, a reduction of GQ would not represent a significant challenge. Improvements in technology for computing and/or IPI NDA measurement performance mean that a GQ of 2kg is potentially feasible. A reduction in timeliness period, with the associated increase in number of IPIs, could represent a greater challenge and certainly real time is not a realistic goal.

A recent paper[13] discussed the importance of process monitoring as an NRTMA prerequisite. It raises in detail a point illustrated by Sellafield Ltd. back in 2001 relating to its Thorp NRTMA system[14]. Underlying the discussion in this paper is the assumption that the IPIs chosen are appropriate. By this the authors mean that there is a demonstrable connection between measurement and plant status at the IPI. As trite as that sounds, without process monitoring this can be almost impossible to demonstrate. As an exemplar, consider the problem – How long should one wait after filling a 100m^3 tank to obtain a pneumacator reading for volume determination?

One IPI a day is sufficient with the proviso that it is the 'right' IPI.

As important as the process monitoring mentioned above is, so is the inclusion of impact assessment on NRTMA in any plant planning/change procedure. An NRTMA system can assess the impact of change in an exactly analogous manner to that described earlier for a new plant, provided

[13] See reference 6

[14] See reference 5

that change is notified expeditiously. The key issue here is one of communication.

The whole concept of NRTMA is based on the ability to model a process in terms of flows and inventory measurements. In fact any analysis based on NRTMA should contain a clear health warning to the effect that the plant must meet the NRTMA plant modelling assumptions. The corollary to this is that, in a no diversion scenario, NRTMA response is a test of business rules and modelling assumptions.

7. Conclusion

It is easy to pigeon hole NRTMA as a tool limited to diversion detection but its application goes well beyond this and, as this paper shows, NRTMA based on Page's Test has significant use at design time before the first brick is laid for the process plant as well as evaluating design change proposals during production.

It has use, not only in materials control applications but also for cost benefit realisation

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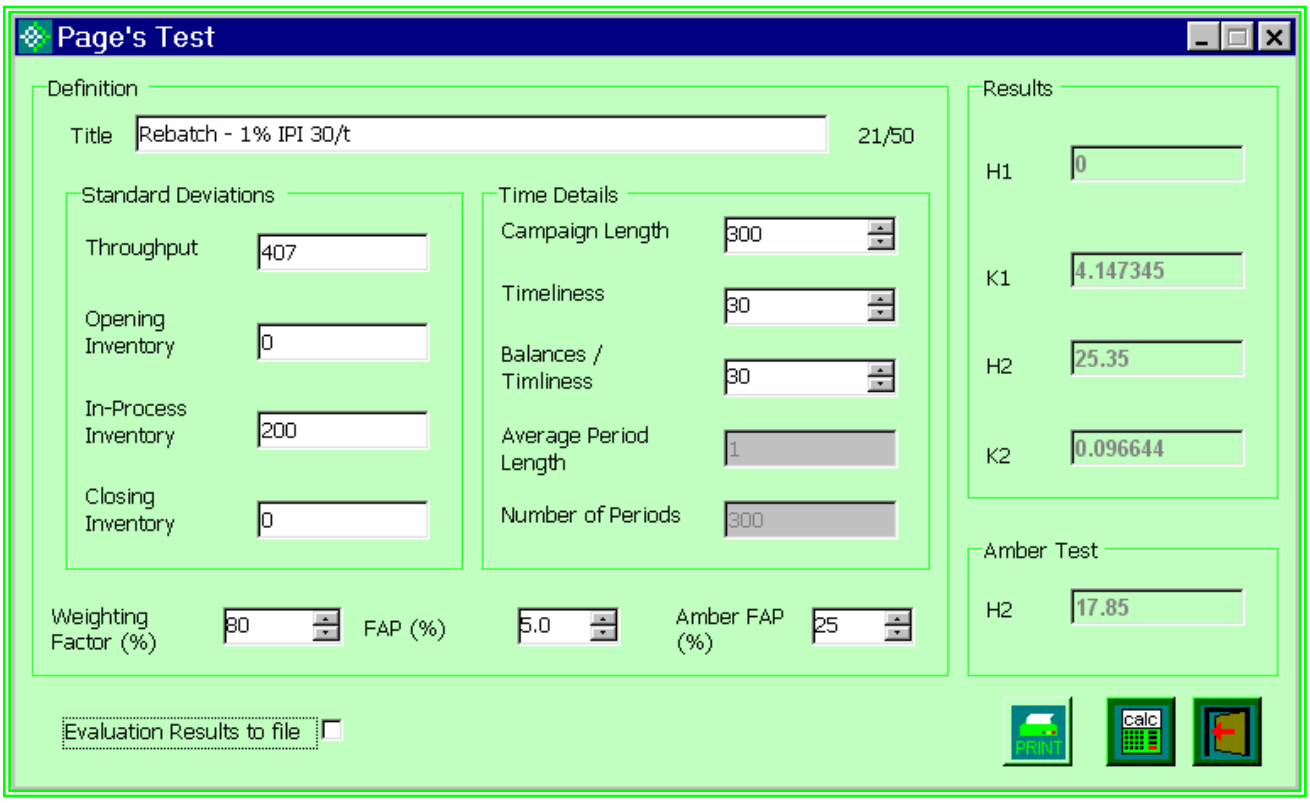
6 G Janssens-Maenhout and L Duchamp, Process Monitoring Appropriate for Near-Real-Time Accountancy, JNMM Spring 2004, Vol XXXIII

Appendix 1 - Creating Page's Tests

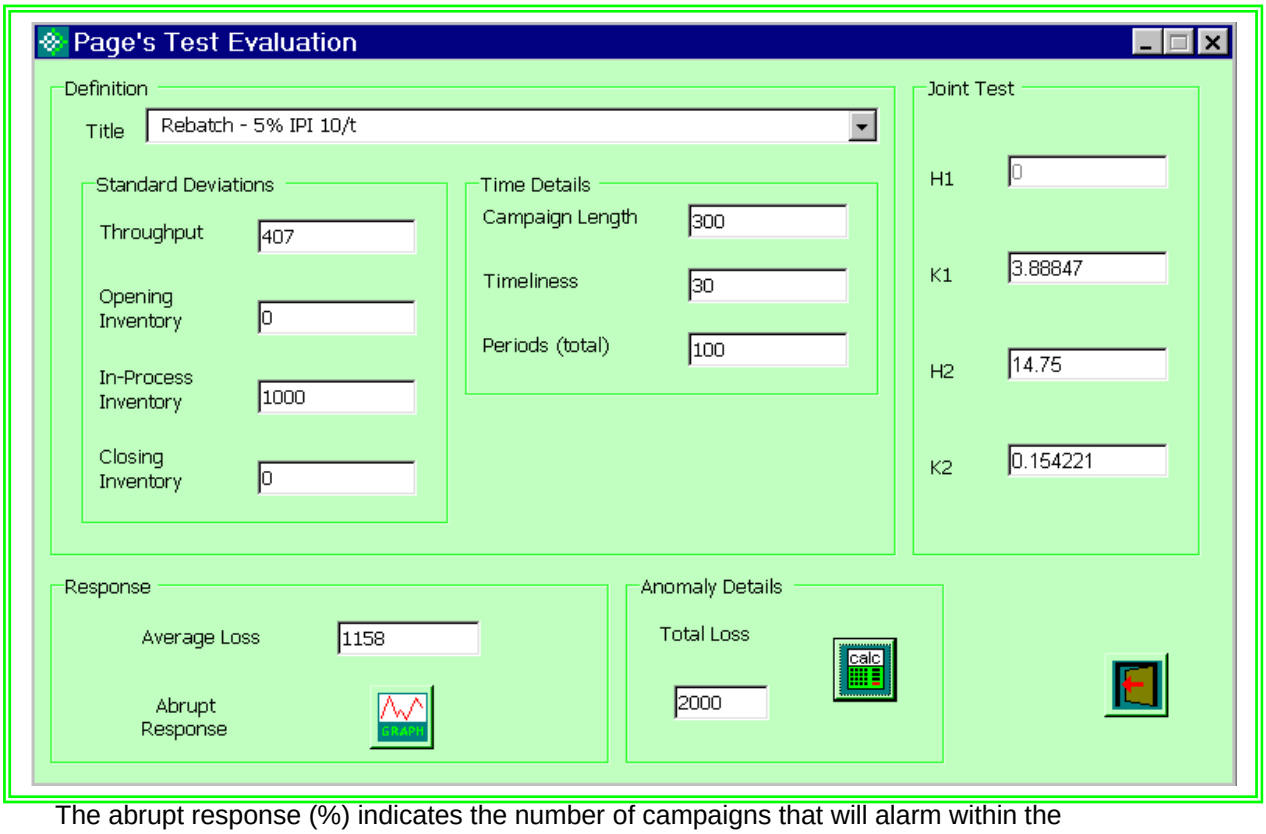
		# Cans	Can Wt.		Can Wt Errors (g) Random	Assay	Assay Error (%) Random	
Make entries here->		**500**	**8000**		**4**	**0.88**	**0.25**	Random
	KMP	n	W	I	Wr	A	Ar	Variance
Receipts	1	500	8000	7040	4	0.88	0.0022	161075.20
Issues	2	50	4000	3520	4	0.88	0.0022	4491.52
							Total (g)	SD
							4000000	**407**



ID	Description	IPI SD	Balances Per Timeliness	Number of Periods	Average Period Length	H1	K1	H2	K2	Amber H2
1	Rebatch - 25% IPI 1/t	5000	1	10	30	0	3.287679	2.15	1.001422	1.35
2	Rebatch - 25% IPI 3/t	5000	3	30	10	0	3.585409	6.05	0.405457	4.1
3	Rebatch - 25% IPI 5/t	5000	5	50	6	0	3.716681	9	0.268135	6.15
4	Rebatch - 25% IPI 10/t	5000	10	100	3	0	3.888443	14.45	0.16065	10
5	Rebatch - 25% IPI 30/t	5000	30	300	1	0	4.147438	27.3	0.082817	19.15
6	Rebatch - 10% IPI 1/t	2000	1	10	30	0	3.288103	3.4	0.596914	2.15
7	Rebatch - 10% IPI 3/t	2000	3	30	10	0	3.585691	7.6	0.278878	5.1
8	Rebatch - 10% IPI 5/t	2000	5	50	6	0	3.716868	10.4	0.205454	7.1
9	Rebatch - 10% IPI 10/t	2000	10	100	3	0	3.888509	15.2	0.144918	10.55
10	Rebatch - 10% IPI 30/t	2000	30	300	1	0	4.147405	26.6	0.087606	18.7
11	Rebatch - 5% IPI 1/t	200	1	10	30	0	3.288258	3.85	0.496134	2.45
12	Rebatch - 5% IPI 3/t	200	3	30	10	0	3.578708	7.7	0.272000	5.2
13	Rebatch - 5% IPI 5/t	200	5	50	6	0	3.716849	10.25	0.211574	7.0
14	Rebatch - 5% IPI 10/t	200	10	100	3	0	3.888470	14.75	0.154221	10.2
15	Rebatch - 5% IPI 30/t	200	30	300	1	0	4.147397	25.8	0.093314	18.15
16	Rebatch - 1% IPI 1/t	200	1	10	30	0	3.288207	3.7	0.528177	2.35
17	Rebatch - 1% IPI 3/t	200	3	30	10	0	3.585629	7.25	0.303968	4.9
18	Rebatch - 1% IPI 5/t	200	5	50	6	0	3.716776	9.7	0.235144	6.65
19	Rebatch - 1% IPI 10/t	200	10	100	3	0	3.888417	14.15	0.167286	9.8
20	Rebatch - 1% IPI 30/t	200	30	300	1	0	4.147345	25.35	0.096644	17.85

Appendix 2 - Page's Test Evaluation Results[15]

ID	IPI SD (%)	Balances Per Timeliness	Number of Periods	Average Period Length	Average Loss (%)	Abrupt Response (%)			
						8000g	*4000g*	*2000g*	*1000g*
1	25	1	10	30	87	**9.0**	*3.6*	*1.6*	*1.3*
2		3	30	10	66	**21.0**	*3.9*	*1.8*	*1.4*
3		5	50	6	57	**34.1**	*5.2*	*2.0*	*1.4*
4		10	100	3	48	**68.7**	*9.1*	*2.4*	*1.4*
5		30	300	1	37	**100.0**	*37.1*	*5.1*	*1.9*
6	10	1	10	30	53	**77.4**	*13.3*	*2.9*	*1.4*
7		3	30	10	40	**99.8**	*28.8*	*4.7*	*1.8*
8		5	50	6	36	**100.0**	*47.3*	*6.6*	*2.1*
9		10	100	3	30	**100.0**	*86.5*	*13.1*	*2.9*
10		30	300	1	23	**100.0**	*100.0*	*53.7*	*6.8*
11	5	1	10	30	36	**100.0**	*73.8*	*11.9*	*2.5*
12		3	30	10	28	**100.0**	*97.1*	*23.8*	*4.1*
13		5	50	6	25	**100.0**	*99.9*	*38.4*	*5.6*
14		10	100	3	22	**100.0**	*100.0*	*74.5*	*10.2*
15		30	300	1	17	**100.0**	*100.0*	*99.9*	*35.0*
16	1	1	10	30	15	**100.0**	*100.0*	*100.0*	*67.0*
17		3	30	10	13	**100.0**	*100.0*	*100.0*	*89.6*
18		5	50	6	12	**100.0**	*100.0*	*100.0*	*96.3*
19		10	100	3	11	**100.0**	*100.0*	*100.0*	*99.5*
20		30	300	1	10	**100.0**	*100.0*	*100.0*	*100.0*



The abrupt response (%) indicates the number of campaigns that will alarm within the timeliness criteria. The size of the loss is based on the concept of a goal quantity of plutonium, which is set at 8kg. The aim is a 95% detection probability. The table also shows the abrupt loss response for losses of 4, 2 and 1kg. It is worth noting that the abrupt loss response is also a function of when through the campaign the loss takes place. Generally the worst performance is at the middle of the campaign and therefore the table reports at that point.

[15] The significance of the Average Loss, a detection measure for protracted loss, and Abrupt response is covered in detail in reference 3

Load-Cell-Based Mass Evaluation Systems Re-assessed on the Basis of URENCO(Capenhurst) Load Cell Data

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Abstract:

URENCO (Capenhurst) have provided extensive load cell data for the purpose of assessing mass evaluation system performance for safeguards purposes. This has enabled the cursory assessment of whether, realistically, station load cell data can form the basis for such a system.

Keywords: monitoring; nuclear; safeguards; centrifuges

1. Introduction

The International Atomic Energy Agency (IAEA) has developed a model safeguards approach that aims to improve upon that established by the Hexapartite Safeguards Project. One of the objectives of this new approach is to confirm that there is no undeclared production of LEU [1]. SAGSI recommended the Secretariat to select the optimal combination of measures to suit each facility, and to provide 'defence in-depth' [1]. This was reiterated in [2]: measures are sought "*to confirm the absence of undeclared production of LEU to enrichment levels not greater than the declared maximum during random inspections complemented by containment and surveillance and monitoring of authenticated load cells"*. The reduced need for verification is often given as a benefit.

A number of papers have mooted the possibility of making more use of station load cell data as a safeguards tool [3-6]. The simplest approach proposed is that of mass balancing [5]. A key conclusion in [5] is that*, "to ensure robust performance it is important that the system is designed for, and assessed with, real data.*" URENCO UK Ltd. have kindly responded by giving access to a large quantity of load cell data pertaining to 2007 operations. This paper describes a cursory investigation that was made with weight data collected from 3 of the more modern units. A number of the older plants are configured in a way that makes it difficult to balance over individual units. Older units can also have steam chests, whose instantaneous weight measurements are more difficult to interpret. The diversion of declared product was 'simulated' by masking data output at one product station, whilst all or part of a cylinder was filled.

In a recent paper [7], URENCO gave their view of what the requirements should be for new inspection techniques. They argue the need for reliability, for acceptable costs and for robustness to false alarms, to operator bypass actions and to sabotage. They also point out that URENCO's enrichment plants contain very sensitive centrifuge technology and potentially, all visitors to the sites contribute to proliferation of sensitive information.

The aim of this paper is to present the results of the cursory investigation, to help decide whether these requirements could be met.

2. Analysis of Capenhurst data

The weight of each cylinder inserted into a station at Capenhurst is measured, hourly, by a 4 load cell arrangement. There is no weighing error when an empty cylinder is placed in a receiving station, because the measured weight is zeroed i.e. its contents weigh nothing. For a feed station, the tare weight is subtracted from the measured weight. It is important to remember that measurements from these weighing systems form part of the unit Nuclear Materials Balance (or so-called Material Unaccounted For, MUF) calculations. Their errors do, already, influence the final result.

Capenhurst stores hourly load cell data as a series of day files, each file pertaining to a different station. This data was combined to form a single, separate year table for each unit for the year 2007. Each station mass history was then analysed, at a relatively cursory level, to identify the periods of time when a station could be deemed to be 'on-line'. The hourly cumulative balance statistic, $S_{[0,k]}$ for each unit was then formed and plotted against time. Integer k is 0 at midnight on New Years Eve., 24 at midnight on January 1st and so on. Statistic $S_{[0,k]}$ can be written as

$$S_{[0,k]} = \left.\begin{matrix}\text{Total}\\ \text{Feed}\end{matrix}\right|_{[0,k]} - \left.\begin{matrix}\text{Total}\\ \text{Products}\end{matrix}\right|_{[0,k]} - \left.\begin{matrix}\text{Total}\\ \text{Tails}\end{matrix}\right|_{[0,k]}$$

where [*0,k*] denotes the interval from 0 to *k*. In an ideal world one would expect: $S_{[0,k]} = S_{[0,k-1]} = \ldots = 0$.

Given all the uncertainties it was quite a surprise to see how the compound effect was relatively linear, for each of the units analysed. To a first approximation:

$$S_{[0,k]} = S_{[0,k-1]} + \Delta = S_{[[0,k-2]]} + 2\Delta = \ldots = k\Delta$$

for considerable parts of the year. An example of this is given in Fig. 1, which shows the cumulative balance for a relatively large unit.

A number of 'sudden' excursions were superimposed on each trend. These excursions could normally be traced to 1 of 4 causes.

1) A failure to identify when a station is on-line correctly, which is hardly surprising given the crudeness of the decision algorithms used. Improved algorithms would correct this. Such incidents were corrected manually.
2) IT issues with transferring load cell data from the stations.
3) Weighing system issues.
4) Operator activities.

In terms of frequencies, cause 1) occurred fairly regularly whilst causes 2) to 4) occurred on average, perhaps once or twice per unit year. That is, very infrequently. The data was adjusted accordingly. Figure 2 shows the cumulative mass for another unit. Note that the scale has been reduced considerably. As a rough guide the long-term bias (drift) equates to a 0.4% error in the measurement of tails weights. Similar trends were also observed in the data collected from the 2 other units that were examined.

The focus of the rest of this paper will be on the data from the second unit (i.e. pertaining to figure 2). This unit has 21 stations, a large proportion of which can be on-line at any one time. On-line weight data from 2 traps are also incorporated. In addition there are a number of other, smaller components, which can contain far less material and hence were ignored in this study. Data from a total of 92 load cells was therefore processed. Unit operation during 2007 was fairly consistent.

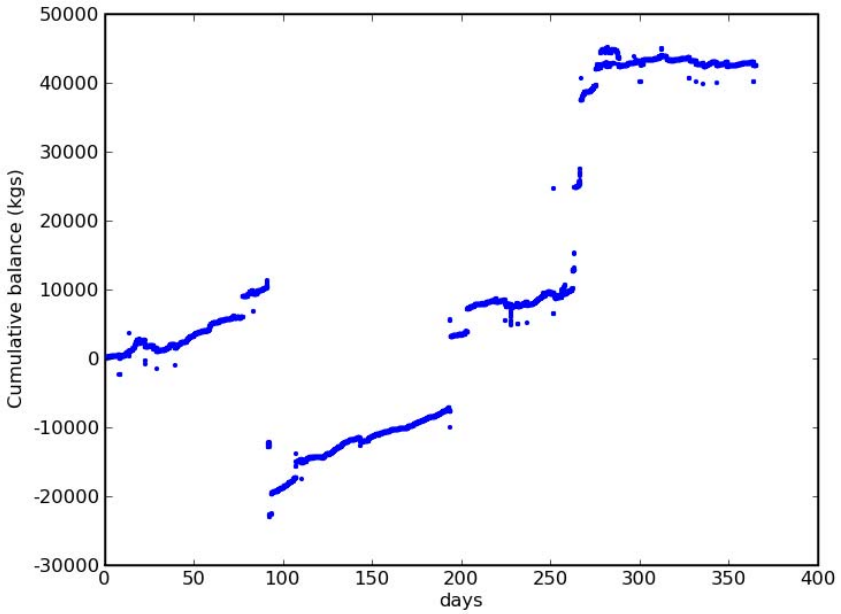

Figure 1: A typical (raw) unit, cumulative balance history

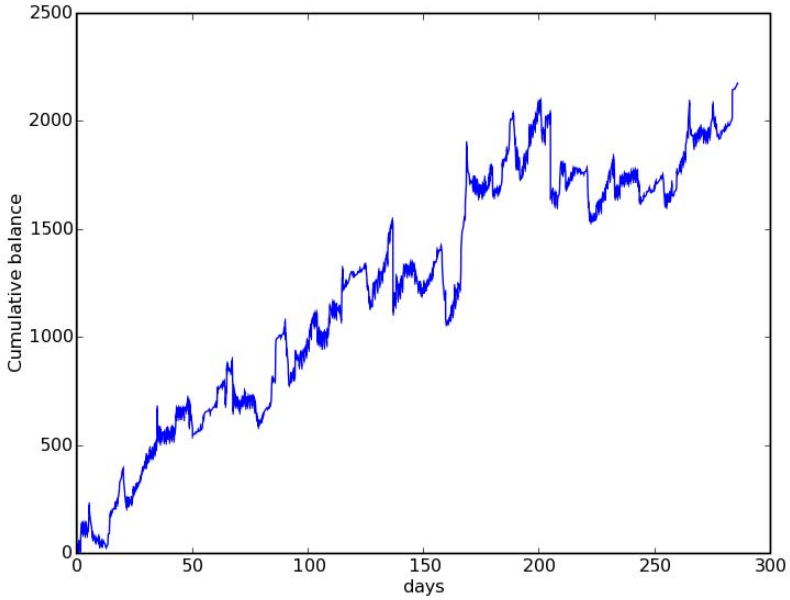

Figure 2: A typical (refined) unit, cumulative balance history

3. Safeguards related performance

'Masked' filling of 1 product cylinder was examined on the unit of figure 2. This filling was simulated by discarding the load cell data that pertained to one product station, whilst its on-line cylinder underwent filling i.e. that station was deemed to be off-line when it was, in fact, on-line. Figure 3 shows the cumulative balance history that resulted. Although the excursion is clearly visible, it might be argued that such an excursion might be confused with one of those seen in figure 1.

A one-sided cusum test:

$$C_k = \max\left(0.0, C_{k-1} + S_{[0,k]} - S_{[0,k-1]} - K\right)$$

was implemented to detect such undeclared removals, with the cusum threshold (*h*) set to infinity to avoid resets, and slope *K* tuned to detect an undeclared fill-rate, of a magnitude similar to that at which a product cylinder is filled normally. To do this, note that a plot of the cumulative balance statistic during this period will have a slope equivalent to the rate of the undeclared fill, provided the plant operates in a perfect, unbiased manner during periods other than during undeclared filling. An investigation of the normal fill rates over the year indicated that, although these were consistently very similar, a different fill rate could arise at times. It was noted that a *K*, chosen somewhere in the range 4 – 10, would align with the slope and hence fail to alarm. A lower value of *K* would detect a removal, but too low a value of *K* would also lead to false alarms, because the test would then be sensitive to the long-term bias observed in all of the cumulative balance histories.

Figure 4 shows plots of the cusum statistic for different values of *K*: the magnitudes of the cusum plots are seen to increase with decreasing *K*. A *K* of 4 with a threshold (*h*) of 500 might be appropriate. Figure 5 shows the same cusum statistics, but with a single product cylinder 'removed'. The removal is clearly visible and would alarm if *K*=4, *h*=500. The ability to detect the undeclared partial filling of a cylinder was then examined by discarding less data. The cusum statistic history pertaining to the undeclared filling of approximately half a cylinder is shown in figure 6. Again it would be detected. Unusually 1 product cylinder was filled at approximately 40% of the normal fill rate for about a quarter of its fill. This gave the opportunity to view the statistic during the undeclared filling of approximately one quarter of a cylinder at 40% fill rate (figure 7). The incident is barely visible at the settings chosen. Reducing the settings would probably necessitate the investigation of false alarms.

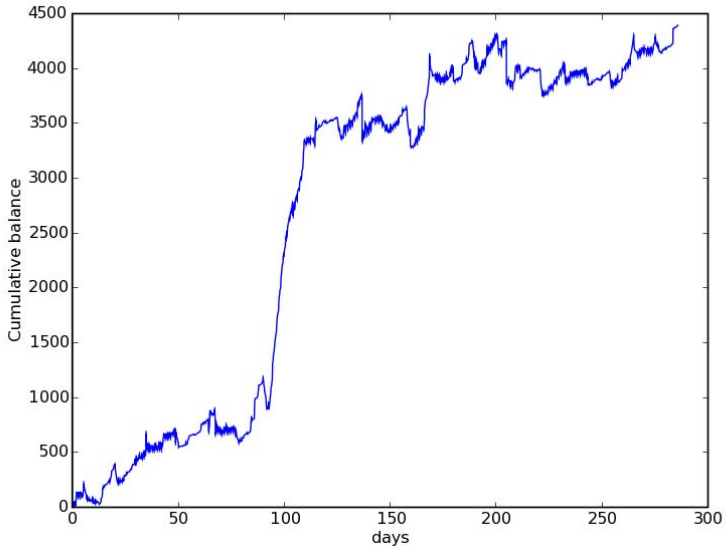

Figure 3: Cumulative balance with an undeclared removal

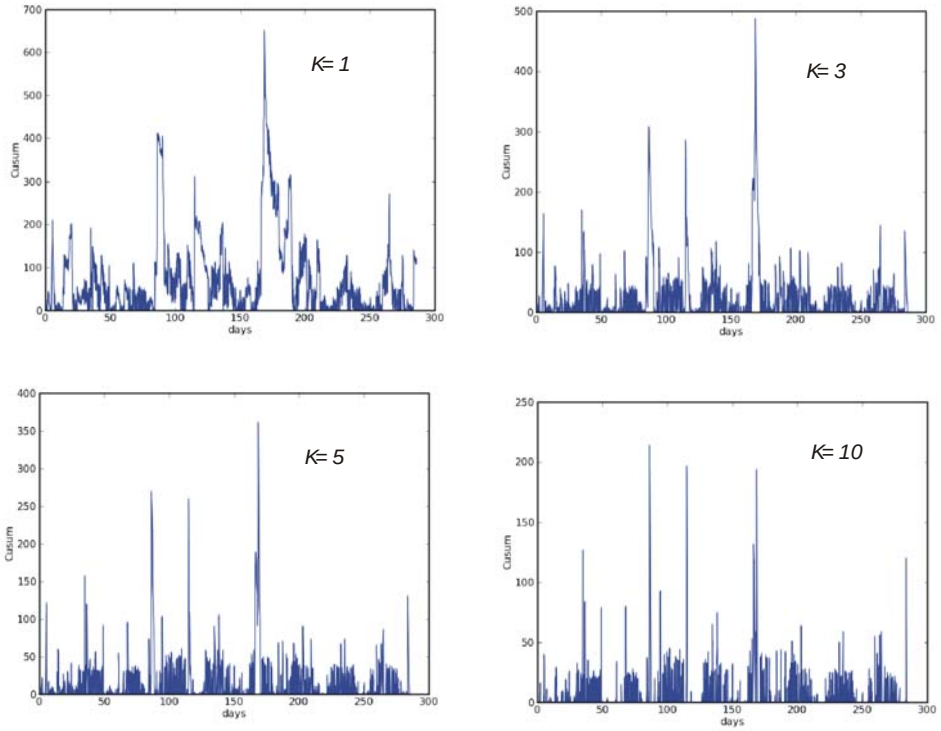

Figure 4: Cusum statistics: *K* = 1, 3, 5 & 10

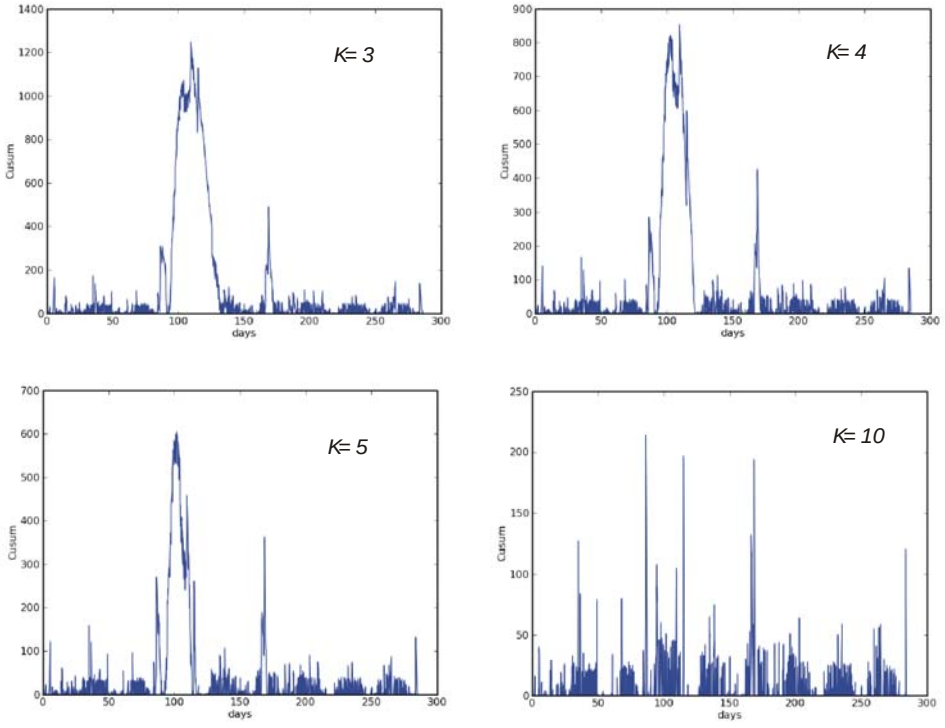

Figure 5: Cusum statistics: undeclared removal A

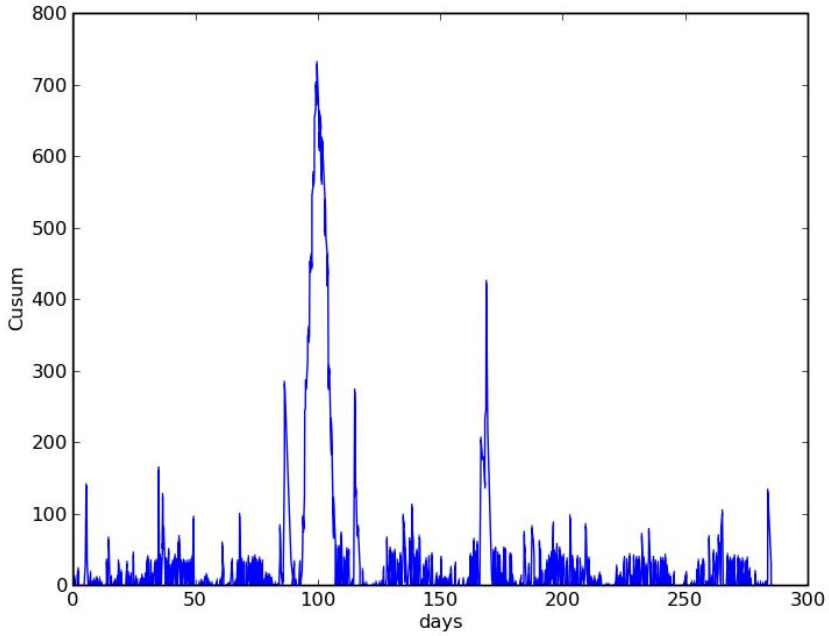

Figure 6: Cusum statistic (*K=4*): half undeclared removal A

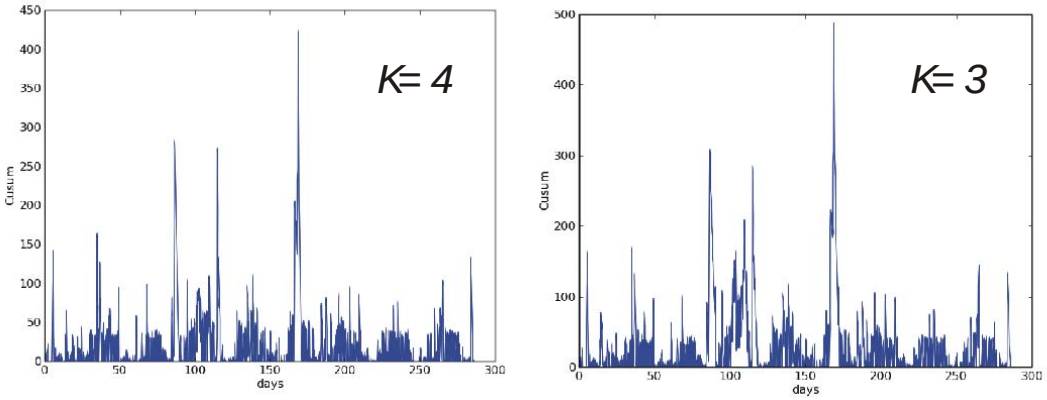

Figure 7: Cusum statistics: 25% full, product cylinder removal at approx. 40% fill rate

4. Discussion

To reiterate from the introduction: the aim of this paper was to help decide whether URENCO's requirements could be met. To reiterate URENCO argue the need for reliability, for acceptable costs and for robustness to false alarms, to operator bypass actions and to sabotage. They also point out that URENCO's enrichment plants contain very sensitive centrifuge technology and potentially, all visitors to the sites contribute to proliferation of sensitive information.

The results of this case study are given in Figures 1-7. Clearly the scales must be viewed with some caution, because these depend on the plant and on the data processing algorithms. It is clear that mass balancing has some potential, because it can detect the undeclared filling of a product cylinder. Whether this potential is sufficient for safeguards purposes is clearly a moot point.

Many graphs were viewed during this study, far more than have been alluded to here. Sudden excursions in cumulative balance histories (like those seen in Figure 1) had to be rationalised in all cases. This raised a concern about data reliability, and hence about intrusiveness. Considerable insight was gained into unit activities, to the extent that the Data Consistency Evaluation approach of [4] could be seen to be a real possibility. Unfortunately it also became clear that this would also raise real concerns with the operator, especially in relation to the proliferation of sensitive information [7].

Although load cell based approaches increase the power to detect undeclared operations, as it stands they would come at some considerable cost in terms of both inspector and capital resources. Clearly as it stands at present, the approach makes use of operator equipment, which the operator has access to. A separate, authenticated system would cost a great deal of money, and might still be open to interference. At this stage it is unclear as to the inspector involvement needed, because the data analysis techniques used above have been fairly cursory and could be improved considerably.

5. Conclusions

From a purely data interpretation point of view, load cell based approaches appear to have potential, although considerable work would be needed to develop a working system. Considerable work would also be needed to establish their operational role in safeguards. Unfortunately it is clear that costs would be considerable in realising an authenticated, reliable system.

It is also clear that these approaches would necessitate greater insight into unit operations, to obtain trends like those of figure 2 and to investigate statistical-test generated alarms, false or otherwise.

Finally it is suggested that the term 'real-time mass evaluation' be replaced by the term 'regular mass-balancing', because the former gives an impression that decisions can be taken instantaneously. Weights might only be recorded hourly (as opposed to all the time) and more than one balance would be needed to observe the evolution of an incident.

6. Acknowledgements

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Testing Material Balance in the Presence of Hold-up or Bias

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Abstract: This paper describes an application of statistical decision theory to the testing of a nuclear material balance. It is aimed at testing a balance when there is some process hold-up which remains unmeasured or where there may be uncorrected biases in the measurement system. Possible examples occur in the case of reprocessing plants and enrichment facilities. For an incomplete balance, the method requires upper and lower tolerance limits for the net effect of holdup on the material balance. For uncorrected biases the method requires a range of values that can be tolerated for the cumulative effect of bias in the balance. The tolerance limits on hold-up and the tolerated range for the effect of bias, are used to determine the composite null hypothesis for the statistical test. The test approach provides a simple and comprehensible management of false alarm risk and is numerically easy to apply. The paper describes the numerical solution for choosing an acceptance region for the balance in terms of the tolerance limits for hold-up and bias, the material balance measurement error standard deviation and the risk aversion parameter.

Keywords: material balance test, decision theory, hold-up, bias.

1. Introduction

In a facility processing nuclear material, assessing the material balance (MUF) at regular intervals is an essential part of assuring that all material is accounted for. The MUF is treated as a Gaussian random variable and the measurement error standard deviation of MUF is denoted σ_{MUF} . This standard deviation reflects the contribution to MUF of those 'zero-mean" components of measurement error that vary under measurement replication. It takes no account of possible error components that remain constant under replication (i.e. biases). Provided there are no biases in measurement procedures and provided the accounting information includes all the material relevant to the balance, the operator's MUF is "an observation" on a probability distribution with zero expected value and with standard deviation given by σ_{MUF} .

This paper considers the balance problem when the situation is not as simple as that described above. Here we consider the problem of assessing the acceptability of a material balance when some small amount of well identified material has not been measured e.g. in process area. In such situations, the balance must be assessed for acceptability even though unmeasured hold-up is not included in the balance computation and hence the balance is an **incomplete balance**. This incomplete balance is the sum of a mean value (not necessarily zero because the hold-up is not accounted for) plus the accumulation of measurement error incorporated in the accounting values of the material that has been included in the balance.

In many cases it is possible to establish upper and lower tolerance limits for the effect of any hold-up that has not been included in the balance. Such limits are not an estimate of what is contained in the hold-up instead they are a statement of what values are to be considered acceptable or tolerable. These limits can be derived taking account of the processing history generating the hold-up. These may then determine what are acceptable values for the mean value of MUF and this range becomes a **composite null hypothesis** for testing the incomplete material balance.

The concept of composite null hypothesis has another application in the assessment of nuclear material balances. Some facility measurement methods may create small measurement biases even though the mass values have been generated by correctly applied procedures. The measurement specialists can be aware that such biases may exist and that their existence is difficult to establish. They may consider that the possible existence of a small bias should be allowed for in assessing the material balance. This can be achieved by using a composite null hypothesis when it is possible to establish a range of values that can be tolerated for the cumulative effect of bias in the material balance.

The presence of bias or hold-up contributions does not affect the definition of σ_{MUF} as the standard deviation of the measurement error in the balance. Again σ_{MUF} represents the cumulative contribution of those error components that have zero-mean and vary under replication. In this report we consider that σ_{MUF} is a known value. The methods used to compute σ_{MUF} for practical situations are described in [1], [2] and [3].

Once a composite null hypothesis is decided and σ_{MUF} has been computed, the MUF is accepted or rejected taking account of the management or inspector's aversion to false alarms. This aversion is represented as a requirement to choose the statistical test rule so that the maximum false alarm probability is equal to a desired target value. The accepted maximum false alarm probability of the inspector will be denoted α_0 . The maximum false alarm probability is referred to as the **size** of the test.

2. Specifying the Composite Null Hypothesis

The facility material balance equation can be written,

$$MUF = \sum_{i=1}^{4} \mathrm{sgn}(i) \sum_{k=1}^{N_i} Z_{ik}$$

where N_i is the number of items referred to in the accounts of the i^{th} MUF component (i.e. BI, R, S, EI; where i=1 means BI , etc.) and Z_{ik} represents the accounting mass value for the k^{th} item in the i^{th} MUF component.

Using M_{ik} to denote the **true mass** of the k^{th} item in the i^{th} MUF component and $L_{ik} = Z_{ik}-M_{ik}$ to denote the **accounting discrepancy** for the item denoted by Z_{ik} , we can write MUF as,

$$MUF = \sum_{i=1}^{4} \operatorname{sgn}(i) \sum_{k=1}^{N_i} M_{ik} + \sum_{i=1}^{4} \operatorname{sgn}(i) \sum_{k=1}^{N_i} L_{ik}$$

where the first term (which contains only the true mass values), is the true material balance and will be denoted $\mathrm{MUF_{TRUE}}$. The second term represents the accumulated discrepancy in the balance and will be denoted $\mathrm{L_{MUF}}$. This notation can be used to write,

$$MUF = \mathrm{MUF_{TRUE}} + \mathrm{L_{MUF}}$$

This **material balance identity** expresses the relationship between any set of balance accounts and a physically existing sets of items referred to by the accounts.

2.1. Tolerance Limits for the Balance Effect of Hold-up Amounts

We now consider the example of an accountancy balance that aims at providing MUF for some MBA in which some material is omitted from the balance account. Suppose that $\mathrm{M_{BI\text{-}holdup}}$ and $\mathrm{M_{EI\text{-}holdup}}$ represent the true masses of material unaccounted for in BI and EI. We write the balance identity for the incomplete balance as,

$$MUF^{*} = \sum_{i=1}^{4} \operatorname{sgn}(i) \sum_{k=1}^{N_i} M_{ik} + \sum_{i=1}^{4} \operatorname{sgn}(i) \sum_{k=1}^{N_i} L_{ik}$$

where now the summation excludes the hold-up in BI and EI and * denotes the incomplete nature of the balance. We also have the compact version,

$$MUF^{*} = \mathrm{MUF^{*}_{TRUE}} + \mathrm{L^{*}_{MUF}}$$

Both $\mathrm{M_{BI\text{-}holdup}}$ and $\mathrm{M_{EI\text{-}holdup}}$ are left out of $\mathrm{MUF^{*}_{TRUE}}$. Suppose now that they are the only omissions. If they were included (completing the sets of material for both BI and EI) it would give a $\mathrm{MUF_{TRUE}}$ = 0. Hence we have,

$$\mathrm{MUF^{*}_{TRUE}} + \mathrm{M_{BI,holdup}} - \mathrm{M_{EI,holdup}} = 0$$

we then have,

$$\mathrm{MUF^{*}_{TRUE}} = \mathrm{M_{EI,holdup}} - \mathrm{M_{BI,holdup}}$$

If we have upper and lower bounds for what are tolerable values for each of $\mathrm{M_{BI\text{-}holdup}}$ and $\mathrm{M_{EI\text{-}holdup}}$, we can then define,

$$\text{Upper bound of } \mathrm{MUF}^{*}_{\mathrm{TRUE}} = \text{Upper bound of } \mathrm{M}_{\mathrm{EI,holdup}} - \text{Lower bound of } \mathrm{M}_{\mathrm{BI,holdup}}$$

and similarly,

$$\text{Lower bound of } \mathrm{MUF}^{*}_{\mathrm{TRUE}} = \text{Lower bound of } \mathrm{M}_{\mathrm{EI,holdup}} - \text{Upper bound of } \mathrm{M}_{\mathrm{BI,holdup}}$$

Such upper and lower bounds for $\mathrm{MUF}^{*}_{\mathrm{TRUE}}$ will be denoted $\left[a\,\sigma_{\mathrm{MUF}},\ b\,\sigma_{\mathrm{MUF}} \right]$ where a and b may be positive or negative and $a \le b$. The upper and lower bounds are in mass units and have no relationship to standard deviations. It is convenient however to express them in units of σ_{MUF} as this will make the statistical test formulae look simpler. From the point of assessing the incomplete balance, we will now consider that having $\mathrm{MUF}^{*}_{\mathrm{TRUE}}$ contained in $\left[a\,\sigma_{\mathrm{MUF}},\ b\,\sigma_{\mathrm{MUF}} \right]$ is acceptable and this interval is the **composite null hypothesis** for the test.

Note also that many situations of hold-up may allow a more effective creation of tolerance limits. When two hold-up amounts, that are of opposite sign in $\mathrm{MUF}^{*}_{\mathrm{TRUE}}$, refer to the same process location (e.g. a tank heel), it may be easier to assign tolerance limits directly to the difference of the amounts. If there are several such pairs of hold-up differences, the separate tolerance intervals for each pair can then be integrated to provide limits for $\mathrm{MUF}^{*}_{\mathrm{TRUE}}$.

2.2 Tolerance Limits for the Balance Effect of Uncorrected Bias

Here we consider a set of accounts with biases in some measurements. We return again to the balance identity,

$$MUF = \sum_{i=1}^{4} \operatorname{sgn}(i) \sum_{k=1}^{N_i} M_{ik} + \sum_{i=1}^{4} \operatorname{sgn}(i) \sum_{k=1}^{N_i} L_{ik}$$

Now we assume that the mass values have been generated by correctly applied procedures but that some measurement methods create small non-zero measurement biases[1]. This means that some L_{ik} are made up of two components. One component will remain constant even if the facility measurements were repeated. This component does not have a probability distribution and is called a **measurement bias**. The second component is a measurement variation whose

[1] Bias is the replication mean of the measurement error.

value would change if the measurement (or its calibration) were repeated. This component has a probability distribution which is incorporated in the computation of σ_{MUF} . The mean of this component is zero.

To make a formal representation of measurement bias we introduce the notation

$$\mathrm{L}_{ik} = \mathrm{B}_{ik} + \eta_{ik}$$

where B_{ik} denotes the bias component of L_{ik} and η_{ik} denotes the component of L_{ik} having a probability distribution. The accumulated discrepancy $\mathrm{L}_{\mathrm{MUF}}$ can be written,

$$\mathrm{L}_{\mathrm{MUF}} = \mathrm{B}_{\mathrm{MUF}} + \eta_{MUF}$$

and the material balance identity can now be written,

$$MUF = \mathrm{MUF}_{\mathrm{TRUE}} + \mathrm{B}_{\mathrm{MUF}} + \eta_{\mathrm{MUF}}$$

where $\mathrm{B}_{\mathrm{MUF}}$ is the net effect of bias and η_{MUF} is the cumulative error component having a probability distribution with zero mean. In this situation σ_{MUF} is just σ_{η} and the mean value of MUF is $\mathrm{MUF}_{\mathrm{TRUE}} + \mathrm{B}_{\mathrm{MUF}}$. If we are discussing a complete balance (i.e. all material is accounted for), the mean of MUF is $\mathrm{B}_{\mathrm{MUF}}$.

To allow for possible bias in the assessment of the balance, it is necessary to provide lower and upper limits for tolerated values of $\mathrm{B}_{\mathrm{MUF}}$. Given such limits, the testing of the complete balance can be carried out using a composite null hypothesis. As an example, we can consider the case of a single potential source of bias. Suppose that the source of bias refers to material that is input to the process. Let Z kgs denote the accountancy value for the total amount of such material in BI and R that is input to the process during the balance period. Suppose that the measurement specialist considers that the measurement method can have a bias per unit mass with a value between λ_1 Kgs and λ_2 Kgs per unit mass.

Suppose we can say that the accountancy value Z may suffer from a tolerable bias contribution λM where $\lambda \in [\lambda_1, \lambda_2]$ and where M is the true value of the material referred to by Z. The limits for tolerable bias contribution in this balance, can therefore be estimated by the lower and upper values $\frac{\lambda_1 Z}{1+\lambda_1}$ Kg and $\frac{\lambda_2 Z}{1+\lambda_2}$. In the simple case where there is only one source of bias being considered, these are then the limits for tolerated values of $\mathrm{B}_{\mathrm{MUF}}$. For application in the formula for tests, these limits must be used to generate the values **a** and **b** for the composite null hypothesis in the notation $\left[a\,\sigma_{\mathrm{MUF}},\ b\,\sigma_{\mathrm{MUF}} \right]$.

Many situations will be more complex than this very simple case. Any accountancy mass value Z

will be a product of bulk determinations (e.g. mass or volume measurement), metal factors (U or Pu) and in the case of ^{235}U, an enrichment determination. Each of the measurement methods used for bulk, metal concentration and enrichment, will have its own potential sources of bias. A bias in one of these methods will transmit into the accounts in function of which particular mass values in the accounts have been determined using that method. For each source of bias being considered, the accountant will have to propagate its effect in the accounts to determine the range of its potential contribution to $\mathrm{B}_{\mathrm{MUF}}$. In a complex application, there may be several relevant sources of potential bias. In that case, each source of bias will generate a range of possible contribution to $\mathrm{B}_{\mathrm{MUF}}$. Taking account of all of these, the global limits for $\mathrm{B}_{\mathrm{MUF}}$ will be computed exactly as when there were several hold-up contributions. In other words the global lower limit will be computed from the combination of source values that gives the smallest lower limit. Similarly the global upper limit will be computed from the combination of source values that gives the largest lower limit.

We see that a composite null hypothesis might be used in two kinds of situation i.e. when there is unmeasured hold-up which is ignored in computing the balance, and when the values for some material in the accounts may incorporate uncorrected biases. Sometimes a composite null hypothesis can be a combination of tolerance criteria for both types of contribution. In this case the upper and lower bounds for the test are the combination of the separately determined upper and lower bounds for $\mathrm{B}_{\mathrm{MUF}}$ and $\mathrm{MUF}_{\mathrm{TRUE}}$. Whatever the case, the upper and lower bounds for the mean value of MUF will be denoted $\left[a\,\sigma_{\mathrm{MUF}},\ b\,\sigma_{\mathrm{MUF}} \right]$.

3 Carrying out the Test of a Composite Hypothesis

Following what has been said earlier about notation, the closed interval $\left[a\,\sigma_{\mathrm{MUF}},\ b\,\sigma_{\mathrm{MUF}} \right]$ where $a \le b$, denotes the range of acceptable mean values of MUF or MUF* i.e. the null hypothesis for the statistical test. The null hypothesis is called **composite** when $a < b$ and is called **simple** when $a = b$. In what follows we will assume that the inspector has a probability value α_0 which is the **largest** false alarm probability that he wishes to tolerate. He wishes therefore to choose a test, whose largest false alarm probability, over the range of null hypothesis values $\left[a\,\sigma_{\mathrm{MUF}},\ b\,\sigma_{\mathrm{MUF}} \right]$, will be α_0.

The acceptance region for the test is denoted $\left[K_1\,\sigma_{\mathrm{MUF}}, K_2\,\sigma_{\mathrm{MUF}} \right]$ where K_1 and K_2 must be computed numerically for the specific values a, b and α_0. These values are computed as $K_1 = K_1^*$ and $K_2 = a + b - K_1^*$ where K_1^* is the solution of

$$\Phi\left(K_1 - a\right) + \Phi\left(K_1 - b\right) = \alpha_0$$

and Φ is the standardised Gaussian distribution function

$$\Phi(K) = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^{K} e^{-\frac{1}{2}u^2} du \quad .$$

This equation for K_1^* is readily solvable (numerically) and hence the method is easy to apply given specific values for a, b and α_0. The inspector will solve the above equation for K_1^* and will then use a decision rule of the form,

" Accept the balance if and only if $K_1^*\sigma_{\text{MUF}} \le MUF^* \le \left(a+b-K_1^*\right)\sigma_{\text{MUF}}$ "

4. Numerical Solutions and their Properties

Writing $y = K_1 - a$ in the equation for K_1^*, we have $\Phi(y) + \Phi\left(y-(b-a)\right) = \alpha_0$. This shows that the value of $K_1^* - a$ is determined only by the values of $b-a$ and α_0.

Since by definition the acceptance region satisfies $K_2\left(K_1^*\right) = a+b-K_1^*$ we have $K_2\left(K_1^*\right) - b = a - K_1^*$. The distance between K_1^* and a is the same as the distance between $K_2\left(K_1^*\right)$ and b. In addition, $K_1^* < a$ if and only if $K_2\left(K_1^*\right) > b$. Note also that if $\alpha_0 < 0.5$ as it usually is in real applications, then $K_1^* < a$.

In this section we look at examples of balance acceptance regions generated by different values of b and α_0. For simplicity we have set $a = 0$ in all examples and hence $b > 0$ also represents the range $b-a$ of the composite null hypothesis. The values for K_1^* and K_2^* in these tables are computed by solving the equation $\alpha_0 = \Phi\left(K - b\right) + \Phi\left(K\right)$ for K_1^* and then putting $K_2^*\left(K_1^*\right) = a+b-K_1^*$. In Tables 1 – 4 below, the simple null hypothesis is represented by the case $b = 0.0$.

Table 1 $\alpha_0 = 0.05$						
Value of 'b' →	0,0000	0,2500	0,5000	1,0000	2,0000	3,0000
$K_1^* \rightarrow$	-1,9600	-1,8502	-1,7697	-1,6815	-1,6461	-1,6449
$K_2^* = a+b-K_1^*$	1,9600	2,1002	2,2697	2,6815	3,6461	4,6449

Table 2 $\alpha_0 = 0.02$						
Value of 'b' →	0,0000	0,2500	0,5000	1,0000	2,0000	3,0000
$K_1^* \rightarrow$	-2,3263	-2,2193	-2,1462	-2,0759	-2,0543	-2,0538
$K_2^* = a + b - K_1^*$	2,3263	2,4693	2,6462	3,0759	4,0543	5,0538

Table 3 $\alpha_0 = 0.01$						
Value of 'b' →	0,0000	0,2500	0,5000	1,0000	2,0000	3,0000
$K_1^* \rightarrow$	-2,5758	-2,4707	-2,4022	-2,3422	-2,3266	-2,3263
$K_2^* = a + b - K_1^*$	2,5758	2,7207	2,9022	3,3422	4,3266	5,3263

Table 4 $\alpha_0 = 0.005$						
Value of 'b' →	0,0000	0,2500	0,5000	1,0000	2,0000	3,0000
$K_1^* \rightarrow$	-2,8070	-2,7036	-2,6393	-2,5876	-2,5760	-2,5758
$K_2^* = a + b - K_1^*$	2,8070	2,9536	3,1393	3,5876	4,5760	5,5758

These examples illustrate a number of general features. The principal feature is that for a fixed value of α_0, the magnitude of $K_1^* - a$ shrinks as the range $b - a$ of the composite null hypothesis increases /4/. As well as this, for $a \le b < +\infty$ we have that,

$$a + \Phi^{-1}\left(\frac{\alpha_0}{2}\right) \le K_1^* < a + \Phi^{-1}\left(\alpha_0\right).$$

For the simple null hypothesis $b = a$, the value of K_1^* is $a + \Phi^{-1}\left(\frac{\alpha_0}{2}\right)$ i.e. the method gives the same acceptance region as the traditional two sided test of the same size. The detection power of these tests is described in detail in [4].

5. The Decision Theory Formulation

In the description until now, these tests have been motivated in terms of looking for a symmetric acceptance region $\left(K_1^*, a+b-K_1^*\right)$ for a specific null hypothesis $\left[a\sigma, b\sigma\right]$ and a desired test size α_0. The same tests however can be motivated by a decision theory formulation in which inspector's risk aversion is embodied in a specific loss function. The same tests emerge as the minimax tests of this formulation [4].

In the decision theory formulation the inspector's risk aversion is represented in terms of two penalty costs i.e. $p_1 > 0$ being the penalty associated with Type 1 error (i.e. cost of false alarm) and $p_2 > 0$ is the penalty associated with Type 2 error (i.e. cost of non detection). It is shown in [4] that the use of the minimax approach involving the choice of values for p_1 and p_2 is equivalent to choosing a value for α_0 and choosing the symmetric rule $\left(K_1^*, a+b-K_1^*\right)$. In the minimax approach, K_1^* is characterised by the equation,

$$\Phi\left(K_1 - a\right) + \Phi\left(K_1 - b\right) = \frac{p_2}{p_1 + p_2}$$

along with the equation $K_2^* = a + b - K_1^*$ as before. The link between these two equivalent formulations is provided by $\alpha_0 = \frac{p_2}{p_1 + p_2}$.

Hence these tests have a dual interpretation either as being derived from a choice of the parameter values for the loss function of the decision maker (operator or inspector) or equivalently as a specific approach to the management of false alarm risk. The formulation emphasizing the maximum false alarm probability is more appealing to safeguards accountants who will have less difficulty in choosing a value for α_0 (a maximum false alarm probability they are willing to tolerate) than in imaging values for p_1 and p_2. The decision theory formulation, the derivation of minimax tests and the equivalence of other formulations is described at length in [4].

6. Different Approaches to Balance Testing

In some earlier approaches [2], [3], testing of a material balance has been presented as a one sided test of a null hypothesis for the mean of MUF.

In [2] two approaches are suggested. The first of these (sect 7.1) is a one sided test of a composite null hypothesis E(MUF) ≤ M_0. Given a value for M_0 the rejection threshold is determined by a desired test size for the composite null hypothesis. In this first approach σ_{MUF} is computed only from measurement error contributions. The second approach in [2] (sect 7.4) brings hold-up into the statistical model by treating the balance effect of hold-up as a random variable with a non-zero mean E(Hold-up) and a standard deviation. For the proposed MUF test, σ_{MUF} incorporates the contribution of hold-up variance along with the measurement error contributions. Again a one sided test is proposed and the rejection threshold is determined by a desired false alarm probability for E(MUF) = E(Hold-up). The difficulties raised by trying to apply this approach are discussed later.

In [3] (sect 7.2.2.2) the null hypothesis is simple i.e. E(MUF) = 0.0 and the desired false alarm probability determines the one sided rejection threshold. Here σ_{MUF} is computed only from

measurement error contributions. The method does not address the question of assessing the balance in the presence of tolerable levels of hold-up or bias.

A one-sided test for rejecting a MUF that is too large is aimed at detection of missing material or accountancy problems that suggest missing material. These can include increased unmeasured hold-up, positive measurement bias on process inputs, negative bias on process outputs, clerical errors suggesting missing material or indeed diversion itself. Such situations produce a MUF value that is too large, and this will tend to be detected by such one sided tests.

The one-sided test however is insensitive to the real introduction of new material or to accountancy errors suggesting that new material has been found. It is insensitive to decreased unmeasured hold-up, negative bias on process inputs, positive bias on process outputs, clerical errors suggesting a gain of material or indeed a real gain of material. Such situations tend to produce a MUF value that is too small and the one sided test will tend to ignore this evidence of accountancy problems. To ensure NMAC quality it is necessary to be able to recognise the effect of accountancy errors, hold-up or bias that reduce the value of MUF as well as causes of MUF being too large. Hence a two sided test is more suitable.

Treating hold-up contribution as if it had a probability distribution [2] can have considerable problems in application and in interpretation of the result. A mean and standard deviation for hold-up may make sense if balance periods are repetitions of a similar set of process activities. Even in this case values for mean and standard deviation can only be proposed on the basis of a sufficient number of measurements of hold-up on successive balance periods and such cleanout measurements are costly. If variations in the hold-up will be influenced by the processing history (e.g. changes in batch characteristics, changes in throughput volumes), we cannot assume that any estimated mean and variance are representative of future balance periods when these will not be a sequence of similar situations.

Any uncertainty about the validity of any estimated mean and standard deviation (either from a small sample of campaigns or because of change in the activities of the balance period) may open the door to subjectivity when a MUF test produces a statistically significant result. The test result may be ignored as being “probably due to bad modelling of hold-up effect”, without any rational justification for this conclusion. As a result the statistical test may be seen as failing to take account of the real operating factors and may be discredited as a tool.

Now consider a balance which incorporates a hold-up contribution and whose measurements involve uncorrected bias. If the balance were tested with a simple null hypothesis E(MUF) = 0.0 , the statistical test may say that the material balance is statistically significant. The explanation of this may be nothing more than the fact that there is a failure to take account of some tolerable bias or hold-up effect. This has a disabling effect on the usefulness of such a test. Such tolerable explanations are suspected but not integrated into the statistical test (via a composite null hypothesis) and test alarms may be ignored as being “probably due to bias or hold-up”, without any further justification in terms of quantitative analysis for this conclusion.

A composite null hypothesis is constructed from statements about,

- what values of net hold-up contribution would be tolerated (taking account of the nature and volume of processing which has taken place),
- what values of bias contribution would be tolerated (given the nature of the measurements involved and their role in the accounts).

The important word here is “tolerated” These values are not a guess at what has occurred. They are simply statements about what is acceptable. They specify acceptable situations of hold-up and bias that are justified in terms of the processing during the balance period, the nature of the process and its measurement system. If a significant MUF results from a minimax test based on a composite hypothesis that includes tolerable effects of hold-up and bias, it suggests that an effect has occurred that is not to be tolerated. The cause can be intolerable hold-up or bias,

accountancy errors or loss or gain of material. Each of these potential causes needs to be addressed as possible explanations of a significant result. Using a composite null hypothesis provides a rational basis for treating these aspects of the situation. When there is hold-up and bias but a composite null hypothesis is not used, there is a risk of acceptance of the balance in spite of its statistical significance because of valid criticism of the statistical test formulation. Applying the composite hypothesis approach leaves little leeway for delegitimating the statistical evaluation when the balance test recognises a significant value of MUF.

7. Conclusion

This paper describes the numerical method for choosing the balance acceptance region in terms of a composite null hypothesis, the material balance measurement error standard deviation and the risk aversion parameter. The test is "two sided" in that MUF is rejected if it is too large or if it is too small. Only balance values near to the null hypothesis range are acceptable. The tolerance limits on hold-up and the tolerated range for the effect of bias can be used to determine the composite null hypothesis for the statistical test. For professional NMAC the composite null hypothesis should have a technological and processing justification derived from the specifics of the volume and type of processing, the nature of the hold-up, the nature of the measurements and their role in the accounts.

The statistical approach described here can be a starting point for developing the facility specific material balance tests recommended under [5]. Such tests are a real and recurrent need of facility operators both for their inventory control program (in process areas) and for the closure of MBA balance at PIT. Safeguards authorities get assurance about the NMAC system when they see a transparent method for material balance closure that takes realistic account of operating contingencies in bulk handling facilities.

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Error estimation in nuclear material weighing

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Abstract

Nuclear material quantities located in nuclear plants consist of additions and subtractions of amounts of different types of materials. Most generally, the quantity of nuclear material held is deduced from 3 parameters: a mass (or a volume of product), a concentration of nuclear material in the product considered and an isotopic composition.

Global uncertainties associated to nuclear materials quantities depend of confidence level of results obtained with the measurement of every different parameter. Uncertainties are generally estimated by considering five influencing parameters (ISHIKAWA's rule): the material itself, the measurement system, the applied method, the environmental conditions and the operator.

A practical application of a procedure to be used to cope with weighing errors is presented in the paper. Results obtained with different kinds of nuclear materials, scales and surrounding conditions, are considered.

Keywords: weighing, nuclear material, uncertainty, error, non destructive measurement

1. Introduction

Domestic and international regulations on nuclear materials impose to have rules enabling each operator to permanently know the localization, quality and quantity of these materials. Physical follow-up is based in particular on measurements and analyzes carried out at key measurement points of the processes implemented in the plant and impacting these materials. Three types of measurements are generally performed: a weighing to determine a quantity of product (example: UF_6, uranium and/or plutonium oxides …), a concentration determination to know the quantity of nuclear material (U, Pu) contained in the product and an isotopic measurement. The required performances of the measurement system (method, technical means…) depend on the final objective to reach. These three variables are more or less well-known (accuracy, uncertainty…) and depend in particular of:

- the performances of the equipments employed,
- the calibration conditions (representativeness of the samples used for calibration compared to the real sample measured),
- the checking and using operations.

This paper defines some problems encountered with weighing. It focuses on the different points that the operator has to consider to define and optimise its measurement system. The operator must define in particular:

- the measurand, i.e. the expression and the unit of the result resulting from measurement (for example a mass of product expressed in kg) and the types of products which will be measured (powder, solids, liquid,…),
- the technical specifications required (tolerances, maximum permissible errors (MPE), uncertainties,…).
- the normative and contextual constraints.

This reflexion determines the choice of the measurement system and the corresponding resources to set up.

2. Definitions

Mass – real mass: the mass (or real mass) m of an object is a physical, constant and intrinsic data of this object; it is equal to the product of its density ρ_m by its volume V ($m = \rho_m V$).

Apparent weight - apparent mass: weighings are generally carried out in the atmosphere (air of density ρ_a). The balance measures a force called the apparent weight. This apparent weight is the sum of two opposed forces, a weight mg and a force due to Archimedes's buoyancy $\rho_a Vg$.

Conventional mass: the conventional mass m_c of a body (of (real) mass m and density ρ_m) is the mass of a fictitious standard of density ρ_0 = 8000 kg/m^3 that balances this body under conditions conventionally chosen: an air of density ρ_{a0} = 1,2 kg/m^3 and a reference temperature t= 20 °C.

Body	ρ_m : density (kg/m^3)	V : volume (cm^3)	$\rho_a V$: air buoyancy correction/gravity (g)	m_{app} : apparent mass (g)	m_c : conventional mass (g)	m : (real) mass (g)
Platinum	21500	47	0,056	999,944	1000,094	1000,000
U metal	19000	53	0,063	999,937	1000,087	1000,000
UO_2 sintered pellets	10500	95	0,114	999,886	1000,036	1000,000
Stainless steel	8400	119	0,143	999,857	1000,007	1000,000
Reference	8000	125	0,150	999,850	1000,000	1000,000
Aluminium	2700	370	0,444	999,556	999,706	1000,000
Solid UF_6 (t= 20°C)	5100	196	0,235	999,765	999,915	1000,000
$U0_2$ powder	2000	500	0,600	999,400	999,550	1000,000

Table 1: Calculation of the conventional and apparent masses corresponding to the same (real) mass of 1000 g for various bodies

Error of indication: it is the difference between the indication of the balance and the value of the standard weighted. This error characterizes the accuracy of an instrument of weighing.

Class of a weight: weights (or sets of weights) are defined according to certain metrological requirements intended to maintain the mass values within specified limits. Nine weight classes are defined by the International Organisation of Legal Metrology (OIML): E1, E2, F1, F2, M1, M1-2, M2, M2-3 and M3. Weights of class E1 are the most precise.

Class of a balance: balances are defined according to certain metrological requirements intended to maintain the error within specified limits. Four balance classes are defined (class I, II, III and IV) in legal metrology. Balances of class I are the most precise.

Resolution of a reading d: it is the value expressed in mass units giving the difference, between two consecutive reference marks for an analogical indication, or between two consecutive indications for a numerical indication. For a numerical instrument, d is the quantification step of this instrument.

Checking step e: it is a value expressed in mass units and used to check an instrument according to the legal metrology rules. Its value depends on the balance characteristics. If e is not specified by the manufacturer, people generally chose e = 10 d.

Maximum permissible error MPE: it is the extreme value acceptable of an error defined by specifications, regulation… for a measurement instrument, a standard weight… This error is generally expressed in units of e.

Error of eccentricity: it is given by the different readings for various positions of the same load on the balance pan.

Fidelity: it characterizes the aptitude of a measuring instrument to give very close indications, for the same load weighed several times and in nearly identical conditions.

Accuracy: it characterizes the aptitude of an instrument to give indications free of systematic error.

3. Weighing of a body-expression of the result

We consider a scale previously adjusted with known conditions. I_o is the indication of weighing before deposit of the body and I_{load} is the indication after. The relationship $\Delta I = I_{load} - I_o$ represents the net weighing result.

3.1. (Real) mass

If the balance had errors of indication E_I which were determined by using standards in conformity with R 111 recommendation of the OIML, if the scale is adjusted just before using it, and if the error E_I was corrected before, the (real) mass m of the weighed body is given by:

$$m \approx (\Delta I - E_I)\left[1+\rho_a\left(\frac{1}{\rho_m}-\frac{1}{\rho_0}\right)\right] \approx \Delta I - E_I + \Delta I \cdot \rho_a\left(\frac{1}{\rho_m}-\frac{1}{\rho_0}\right) \approx \Delta I + \Delta I.\rho_a.\left(\frac{1}{\rho_m}-\frac{1}{\rho_0}\right)$$

3.2. Conventional mass

Using the same notations and assumptions as previously, the conventional mass m_c of the weighed body is given by the relationship:

$$m_c \approx (\Delta I - E_I)\left[1+(\rho_a-\rho_{a0})\left(\frac{1}{\rho_m}-\frac{1}{\rho_0}\right)\right] \approx \Delta I - E_I + \Delta I \cdot (\rho_a-\rho_{a0})\left(\frac{1}{\rho_m}-\frac{1}{\rho_0}\right) \approx \Delta I + \Delta I.(\rho_a-\rho_{a0}).\left(\frac{1}{\rho_m}-\frac{1}{\rho_0}\right)$$

4. Influencing factors

The operator must always keep in mind the required objective of the measurement process. Some criteria (performances, uncertainties…) must have been proposed as indicators to demonstrate that the process complies, the necessary controls are determined and the corresponding measurement means to use (method, equipments) are defined.

Different factors influence directly the quality of measurements and the associated uncertainties: the environmental conditions, standard weights used, operator's competence, instruments used, weighing methods implemented (simple or double weighing.). The cause-effect diagram (fig.1), called the Ishikawa's diagram (or 5 M diagram), gathers the principal quantities to consider. Some variables may, of course, appear negligible; they must however be identified and indicated.

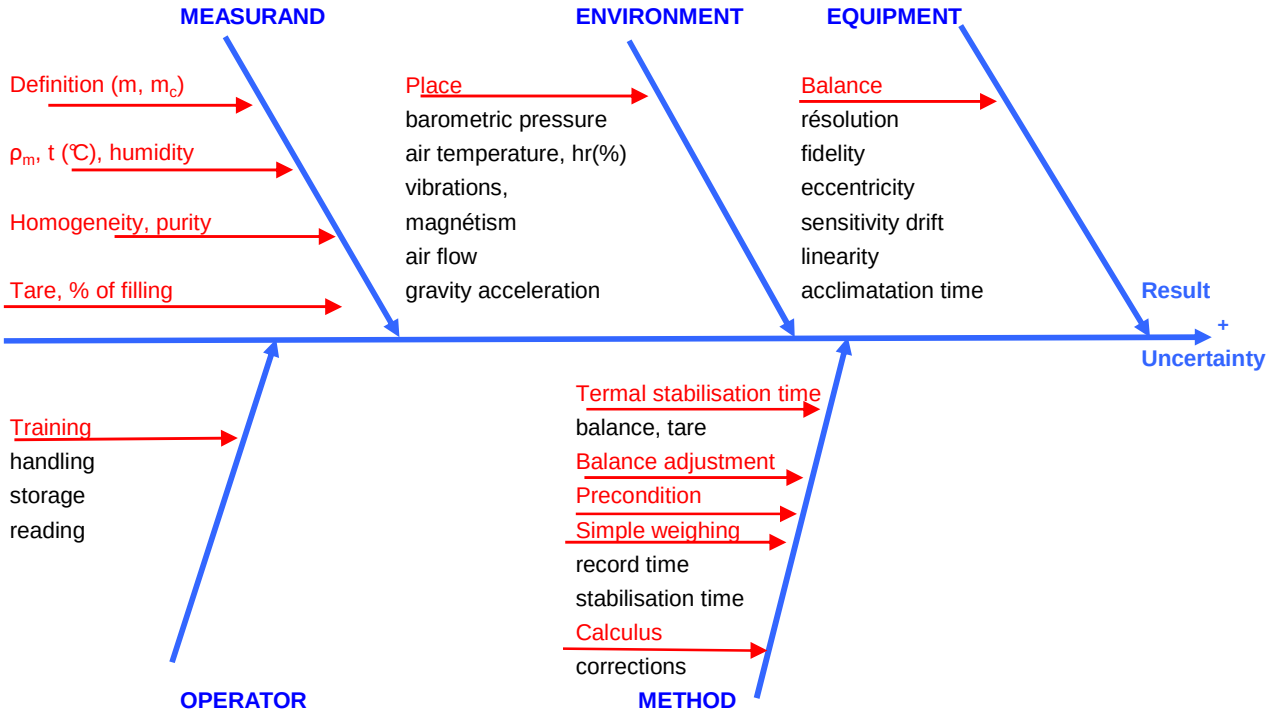


Fig.1: Ishikawa's diagram (or 5 M diagram)

4.1. Measurand

The measurand definition and its characteristics have direct influence on the weighing result and the associated uncertainty. The measurand must be defined carefully, with its unit specified and the most complete possible list of influencing elements.

4.1.1. Definition

According to the measurand wanted (conventional mass or (real) mass), the air buoyancy correction is different and the uncertainty calculation must thus take account of it.

4.1.2. Temperature

A variation of the body temperature causes variations of volume and thus variations of apparent mass. It is necessary to know the cubic dilation coefficient α (expressed in $℃^{-1}$) of the weighed body, to evaluate le real volume at temperature t: $V_t = V_{20}(1+\alpha(t-t_0))$ with $t_0 = 20℃$.

4.1.3. Body density

The product to be weighted can be composed of different bodies having various properties, in particular densities. It is advisable to calculate an equivalent density ρ_{eq} for the whole body (composed of the container, nuclear material, air…) to apply correctly the buoyancy correction. For example a steel container containing material X and air can be seen as a homogeneous body of density ρ_{eq} , mass m_{total} and volume V_{total} with:

$$\frac{1}{\rho_{eq}} = \frac{1}{m_{total}}\left(\frac{m_{steel}}{\rho_{steel}} + \frac{m_{air}}{\rho_{air}} + \frac{m_X}{\rho_X}\right),\ m_{total} = m_{steel} + m_{air} + m_X \text{ and } V_{total} = V_{steel} + V_{air} + V_X$$

The graph hereafter shows the importance of the densities of bodies in weighing. In nuclear material safeguards, three domains are particularly distinguished:

- densities extending from 2 000 to 5 000 kg/m^3; this corresponds to material in liquid form or powders,
- densities close to 11 000 kg/m^3; this essentially corresponds to sintered pellets,
- and densities about 19 000 kg/m^3 corresponding to U and Pu in metal form.

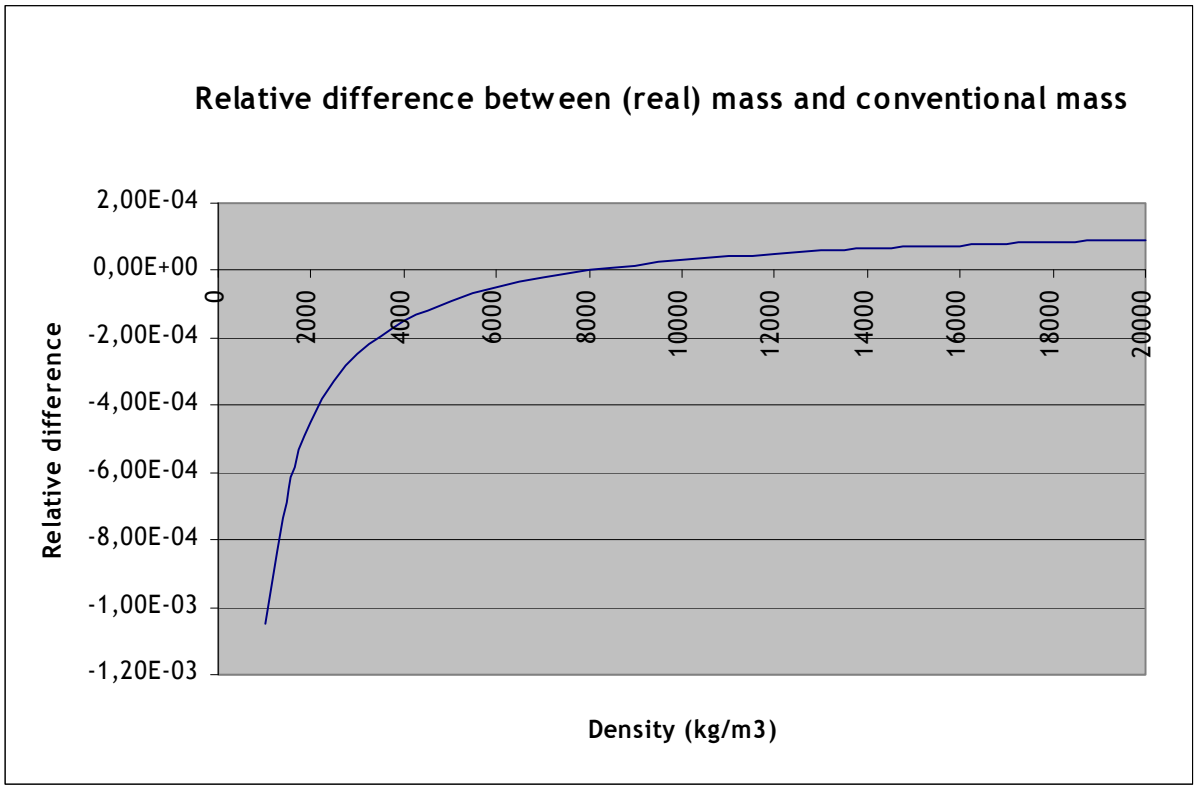


Fig.2: Relative difference between a mass and its conventional mass

So, for example, in a UO_2 fuel fabrication plant, we suppose that uranium in powder form is the input material. This uranium is transformed in sintered pellets to produce the output materials, the fuels assemblies. A relative difference of (real) mass about $3{,}7.10^{-5}$ exists between both material quantities and has to be taken in account to estimate the real quantity of material having passed through the plant. This difference is only due to the difference of densities between both products. This quantity seems to be very small but is not negligible according to the total quantities transformed.

4.1.4. Tare problem

If the container tare intervenes in the final calculation, the corresponding measurand has to be specified. An empty container must be well characterized to avoid any problem due to the mass of possible product remaining inside the container (gas for example). For instance two UF_6 cylinder of internal volume of 1 m^3 (one is filled with air at atmospheric pressure and the other is really empty (because of pumping) and weighed in the same environment (balance, environmental conditions), present a difference in readings of about 1,2 kg.

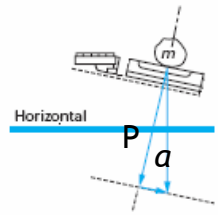
4.2. Operator

The operator's competence can influence the measurement result. Several causes are at the origin of human errors, in particular, the non respect of procedures (for example the acclimatization durations to obtain the thermal stability…), miscalculations, errors in manual records of results…

4.3. Environment

4.3.1. Levelling

A weighing device must always be level (the air bubble must stay in the mark centre). The bubble position must be corrected, if needed, by acting on the adjustable feet. In this example, the balance measures only the P component, perpendicular to the pan. For example, in the case of the IC34000P balance of the laboratory the gradient $tg\alpha \cong \alpha$ must be lower than 8.10^{-4} rd to have a relative error $\Delta m/m$ lower than 3.10^{-5}.



4.3.2. Setting of a weighing device

A setting complies with weighing measurements if:

- the influence of drafts is limited (by moving away the device from doors, ventilations, heating sources and air-conditioning),
- the influence of direct radiations is controlled (by moving balance far from windows),
- the weighing device is placed on a stable and rigid base (no vibration) and is protected from shocks during handling operations,
- the weighing device is protected from static electricity and magnetism problems.

4.3.3. Environmental conditions

In all cases, the weighing device environment (air temperature, barometric pressure, humidity) must be stable. Indeed these parameters directly influence the balance sensor and the air density. The following simplified formula [1] can be used to determine the air density ρ_a (kg/m^3) according to the barometric pressure p (in hPa), temperature t (in °C) and relative humidity of the air h_r (in %):

$$\rho_a = \frac{0{,}34848 \cdot p - (0{,}009 \cdot h_r . \exp(0{,}061 . t)}{273{,}15 + t}$$

This equation has a relative error of 2.10^{-4} for measurements in the range:
900 hPa < p < 1100 hPa, 10 ℃ < t < 30 ℃ and h_r < 80 %.

4.3.4. Local gravity acceleration

Electronic weighing devices measure the force induced by the body weighed but do not determine its mass. The gravity acceleration g depends on location (altitude and latitude). The balance indications thus depend on local g. Carrying out a scale adjustment (internal or external) on site, makes it possible to regulate correctly and automatically the influence of the local gravity. The formula hereafter [3], makes it possible to determine the local g (in m/s²) according to the latitude Φ (in degree) and altitude H (in meter) of the site.

$$g = 9{,}780318.(1 + 0{,}0053024 \sin^2 \Phi - 0{,}0000058 \sin^2 2\Phi) - 0{,}000003085.\mathrm{H}$$

For example, we consider a weighing device adjusted, calibrated and used in Fontenay-aux-Roses (near Paris). It indicates 10 kg for a mass of 10 kg. If this device is transported to Pierrelatte (in the south of France) but is not readjusted on this site, although the same mass of 10 kg is measured, this mass does not generate the same force in both places mentioned, due to the gravity g variations.

We have: $\frac{P_{FAR}}{g_{FAR}} = \frac{P_{Pier}}{g_{Pier}}$ so $P_{Pier} = P_{FAR}\frac{g_{Pier}}{g_{FAR}} = 10000 \times \frac{9{,}8052}{9{,}8084} = 9996.7g$

Fontenay-aux-Roses	Pierrelatte
Latitude Φ = 48°, altitude H = 160 m	Latitude Φ = 44°, altitude H = 53 m
g_{FAR}= 9,8084 m.s^{-2}	g_{Pier} = 9,8052 m.s^{-2}
$P_{FAR} = mg_{FAR}$	$P_{Pier} = mg_{Pier}$
Indication of the balance adjusted in FAR:10000,0 g	Indication of the balance not re-adjusted on site : 9996,7 g

4.4. Equipment

4.4.1. Weighing device

The balance performances affect the uncertainty of the final result. The principal factors to be considered are: the resolution of the scale, its sensitivity, its fidelity, its eccentricity limits and its linearity.
The metrological characteristics of the balance intervene in the evaluation of the instrument uncertainties. Four classes of balances (class I, II, III and IV) are defined according to their performances. Class I corresponds to a high precision apparatus used in a laboratory, class IV corresponds to current appliances. In nuclear field, an operator generally needs a balance of class II for his physical follow-up of nuclear material. The devices of class I are used by laboratories carrying out very precise analysis.

The uncertainty calculus is specific of the different components supposed to be influencing. It depends on the method used for calibration, checking and weighing. Two approaches are presented and compared hereafter “the COFRAC approach” and “the legal metrology approach” to determine the scale uncertainty.

4.4.2. COFRAC approach

In order to accredit a weighing system, according to ISO/IEC 17025 standard, COFRAC (French Committee for Accreditation) has defined [2] specific requirements for the calibration operations and use of the instrumentation. The uncertainty associated with a weighing result is a function of different operations: calibration with standards, conditions of use, COFRAC distinguishes two successive steps to calculate the balance uncertainty.

- The 1rst step is used to determine the error of indication E_I (internal calibration of the balance and the associated and expanded (k=2) uncertainty $U(E_I)$.
- The 2nd step is used to determine the expanded uncertainty U(IP) of the scale.

Steps	Operation	Results obtained	Commentaries
1rst step	Determination of the indication error. Associated uncertainty	Indication error: E_I Expanded (k= 2) uncertainty $U(E_I)$	Operation called calibration
2nd step	Determination of the balance uncertainty	Expanded (k = 2) uncertainty U(IP) of the balance	Exploitation of calibration

The calculation takes into account the reading resolution of the balance, the weighing repeatability, the uncertainties of standards, the effects of eccentric loadings, the temperature, the air density.

The error of indication E_I is given for a load. It is obtained by difference between the result of the simple weighing of a standard load and its certified value. The indication error E_I is given with one or more load values belonging to the domain of use. Loads may be applied with growing and/or decreasing values.

The temperature t influences directly the sensor response according to the load deposited on the balance pan. The relative uncertainty due to the temperature effect on the weighing instrument is given by the relation: $u(t)_{rel} = C. (\Delta t)/\sqrt{3}$ (uniform distribution) with:

- C: variation coefficient of the instrument slope according to the temperature,
- Δt: variation in temperature during calibration (supposed $t_{starting} = 20℃$).

This coefficient C depends on the type of instrument; it is often given by manufacturer. In case of data absence, the table 2 hereafter gives acceptable values.

Maximum number of steps e of the instrument m = Load/e	Maximum variation coefficient versus temperature for approved instruments ($℃^{-1}$)	Maximum variation coefficient versus temperature for other instruments ($℃^{-1}$)
1 000	250×10^{-6}	$2\,500 \times 10^{-6}$
10 000	25×10^{-6}	250×10^{-6}
100 000	5×10^{-6}	50×10^{-6}
>100 000	$1{,}5 \times 10^{-6}$	15×10^{-6}

Table 1: Variation coefficient of the balance sensitivity according to the temperature

Application to the IC34000P balance
This methodology was applied to the IC34000P balance of the laboratory. The COFRAC calculations use linear regression equations resulting from the experiments. Two models giving the expanded uncertainty (k=2) related to the balance itself U(IP) were obtained, each one characterizing a range of use.
$U(IP) = 0.3 + 1.75.10^{-4}.\Delta I$ for the range 0 g – 16 000 g
and $U(IP) = 0.7 + 1.77.10^{-4}.\Delta I$ for the range 16 000 – 34 000 g

4.4.3. Legal metrology approach

The legal metrology approach is simpler. Following an internal or external adjustment of the device on the site, the operator carries out, with standards, a fidelity test (4 to 5 measurements), an eccentricity measurement and a linearity test (on the balance range). The resolution with and without a load intervenes also. The acceptance criteria for these tests are that each difference observed between any result read and the corresponding load used is lower than the balance MPE (maximum permissive error). The balance uncertainty U(IP) is chosen equal to the balance MPE. This variable varies according to the load but is supposed constant over a weighing range.

Different MPEs are found. For example, with the IC34000P balance, a tunnel of errors is obtained: The corresponding MPEs are 1 g for the range 0 – 8 000 g, 2 g for the range 8 000 – 20 000 g and 3 g for the range 20 000 – 34 000 g.

All the results obtained with the IC34000 P balance are presented hereafter. The balance uncertainty calculated with the legal metrology approach applied by the laboratory gives a maximal uncertainty.

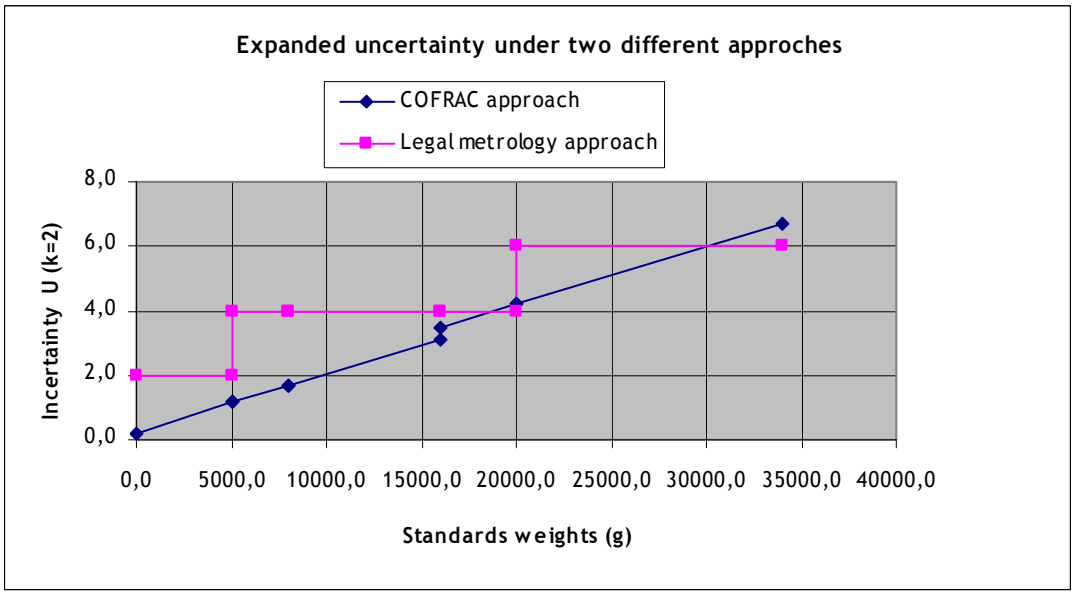


Fig.3: Expanded uncertainty for a IC34000P balance calculated with 2 approaches (COFRAC and legal metrology)

4.4.4. Choice of standards

To adjust, calibrate and check balances, it is necessary to use internal standards correlated to international standards (traceability). The use of magnetized standards, of weights having a thermal equilibrium not reached… are sources of errors to be avoided. The MPE of the standards used must be lower than 1/3 of the balance MPE.

4.4.4.5. Equipment transport

Any transport presents risks for the equipment. It thus should be taken care that the moved balanced will be considered operational only if, before starting measurements, adequate controls (adjustment, tests of eccentricity, repeatability and linearity) are carried out.

4.5. Method

The methods used for calibration and weighing, the measurements number have a direct influence on the weighing result and its associated uncertainty. A Gauss weighing (double weighing) is more precise than a simple weighing (biases are cancelled) but requires two successive operations and standards. The more the calibration and checking conditions are close to the real conditions of use, the less the results have to be corrected and the less important the uncertainty associated with the result is.
Procedures, uncertainty calculations differ according to the selected approach (legal metrology or COFRAC approach). Uncertainties are more or less important and depend on the method used.

5. Conclusions

The weighing is often considered a priori as a very simple operation. It depends on the required objective (performances, needs...). In carrying out a 5 M standard analysis (Ishikawa's diagram) in order to evaluate the different uncertainty sources of the process, the operator realizes that many factors are to be taken into account if a relative uncertainty of about 10^{-5} to 10^{-4} is required for his process control. Three factors have to be considered in priority: the intrinsic performances of the instrument, the environmental conditions of calibration and current use, the physicochemical form of the weighed products.

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SESSION 19

FACILITY SPECIFIC SAFEGUARDS

THE APPLICATION OF NUCLEAR SAFEGUARDS AT THE SELLAFIELD PRODUCT AND RESIDUE STORE (SPRS)

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ABSTRACT

Sellafield Product and Residue Store (SPRS) is a new facility being constructed on the site of Sellafield. The design work commenced in early 2001 and active commissioning is planned to commence in 2010, with first nuclear material arriving in the building at the end of 2010,and the store going into full operation by December 2010. It has been designed for the long term storage of Plutonium product (PuO2) from Thorp and Magnox, MOX residue powder from Sellafield MOX Plant (SMP) as well as pellet, powder or granular PuO2 residues from the older stores on the Sellafield site.

This paper describes the Safeguards Approach to be applied by DG TREN at the Sellafield Product and Residue Store (SPRS). The approach has been developed based upon the requirements for implementing Commission Regulation 302(2005) and the technical measures to be implemented in order to meet Article 77(a) of the Euratom Treaty. The Safeguards Approach developed is in accordance with the internal DG TREN inspection guidelines developed over the last two-three years. Included in the paper will be the techniques that will be applied in the verification of the declared Basic Technical Characteristics when they are declared as well as the impact on operations arising from the application of the safeguards verification activities.

Keywords: Sellafield; plutonium; nuclear safeguards; SPRS (Sellafield Product & Residue Store)

1. INTRODUCTION

1.1. General

The Sellafield Product and Residue Store (SPRS) is a new facility in the course of construction on the site of Sellafield for long term storage of the PuO2 product generated from the reprocessing operations on that site. The design work commenced in early 2001 and active commissioning is planned to commence in 2010, with the first nuclear material arriving in the building at the end of 2010. The store is planned to go into full operation by December 2010. It is being designed for the long term storage of Pu product from Thorp and Magnox, MOX residues in the form of PuO2 powder from SMP as well as pellet, powder or granular PuO2 residues from the older stores on the Sellafield site. The present store has a capacity for 9,600 cans, located in channels that can hold up 15 cans per channel. The store has been constructed in a modular fashion to allow future expansion when the store has run out of capacity.

Discussions on safeguarding this plant commenced soon after the construction in 2004 and there has been an early involvement with DG TREN on the design concept with the design and future operations

teams of this facility. This dialogue has enabled both parties to discuss the impact of safeguards requirements on the proposed operations, the layout of the store, and restrictions arising from any necessary handling operations of the storage containers. DG TREN presented an overall draft safeguards approach which was used to identify the key measurement points and the proposed locations for the different monitoring instruments (cameras, seals, neutron monitors, etc). The dialogue with the design team has enabled DG TREN to settle the boundaries of the proposed Material Balance Areas and to avoid a possible expensive redrafting of suggested surveillance boundary lines.

For safeguards purposes it was agreed that 100% of the cans entering into the store will go through one of the two installed automated Neutron-Gamma measurement station. This operational constraint was necessary as the product and residue material will be delivered from different parts of the Sellafield site. In order to regain the knowledge of the material as it passes into the SPRS store a measurement of the can contents using the CCMs will be performed to facilitate this. The measurement output from the CCMs will be branched so that DG TREN so that the can independently calculate the measurement values for verification purposes. The Can Contents Monitors (CCM) are based upon a passive neutron coincidence counter (PNCC) design with multiplicity analysis together with High Resolution Gamma Spectrometry (HRGS) in order to determine the isotopic composition of the plutonium in each can.

Cans are stored in channels within the store. Once they arrive in these channels their re-verification becomes difficult in practical terms due to the large number of can withdrawals and can reshuffling that would be necessary in order to access specific cans. Therefore multiple Containment and Surveillance (C/S) will be applied to retain the continuity of knowledge of cans and thereby reduce the requirements for subsequent re-verification. There will be a very limited number of re-measurements during the annual Physical Inventory Verification (PIV) as part of the overall assurance scheme.

Sub-perimeters have been set up within the main C/S perimeter to allow greater operational flexibility and to reduce the effects of major C/S failures. These arrangements would allow man-entry to a part of the C/S zone whilst preventing access to the main store. The C/S system is automated as far as is possible to reduce inspection effort.

The proposed lifetime of the plant extending to possibly 2120 means that nuclear material inventory will be in a static state after the initial loading and as such the overall approach to the inspection regime will need to be re-examined to reflect this.

A particular issue with the SPRS plant is the passive cooling of the storage channels. In order to prevent the possible movement of material within the interconnecting air passages upstream and downstream a number of actions have been taken during construction. These include the installation of a number of physical barriers, preventing access via the plenum or air inlets and the use of appropriate technical means to provide assurance that these barriers have not been removed. In addition containment inspections maybe carried out on a short notice basis to confirm that the necessary measures undertaken during the construction phase have not been modified or altered.

2. SAFEGUARDS OBJECTIVES

The objective of the safeguards approach will be the detection of one or more of the following:

- Inconsistencies or shortcomings in the operator's nuclear material accountancy system including obligation accounting.
- Inconsistencies between the Basic Technical Characteristics (BTC) and the physical characteristics of the plant.
- Detection of one missing PuO_2 filled can within 1 material balance period.

3. SAFEGUARDS APPROACH

3.1. General Scheme

To achieve the Safeguards Objectives, the Safeguards Approach is based upon the following:

- An initial verification and re-verification of the Basic Technical Characteristics (BTC).

- The verification of the accountancy and operating data supplied by the operator.
- The physical verification of nuclear material and maintenance of the continuity of knowledge by the application of containment and surveillance.

The proposed safeguards scheme will comprise the following elements:

- An initial verification of the BTC during construction and commissioning, including the verification of the accountancy system, followed by regular subsequent re-verification.
- The verification of receipts and shipments.
- The physical inventory verifications (PIV)
- Interim verifications for flow verification.

3.2. Ventilation Ducting

There is a need for surveillance of the ventilation ducts due to their size and linked to a possible diversion scenario for removal of cans. In order to accommodate this aspect certain controls have been chosen including the use of a 3D laser scanner device, developed by the JRC Ispra, on the grille of the outlet ventilation duct, to be able to detect any form changes over time. The use of the scanner for design re-verification will form part of the annual PIV design verification activities.

3.3. Proposed Safeguards Scheme

The proposed safeguards scheme is based upon two MBAs with one covering the handling area for all nuclear material movements into the building and the second one the store itself. All nuclear material movements into the store will be through one of the two installed CCMs supported by a combination of cameras, seals, and monitors providing a sound surveillance boundary.

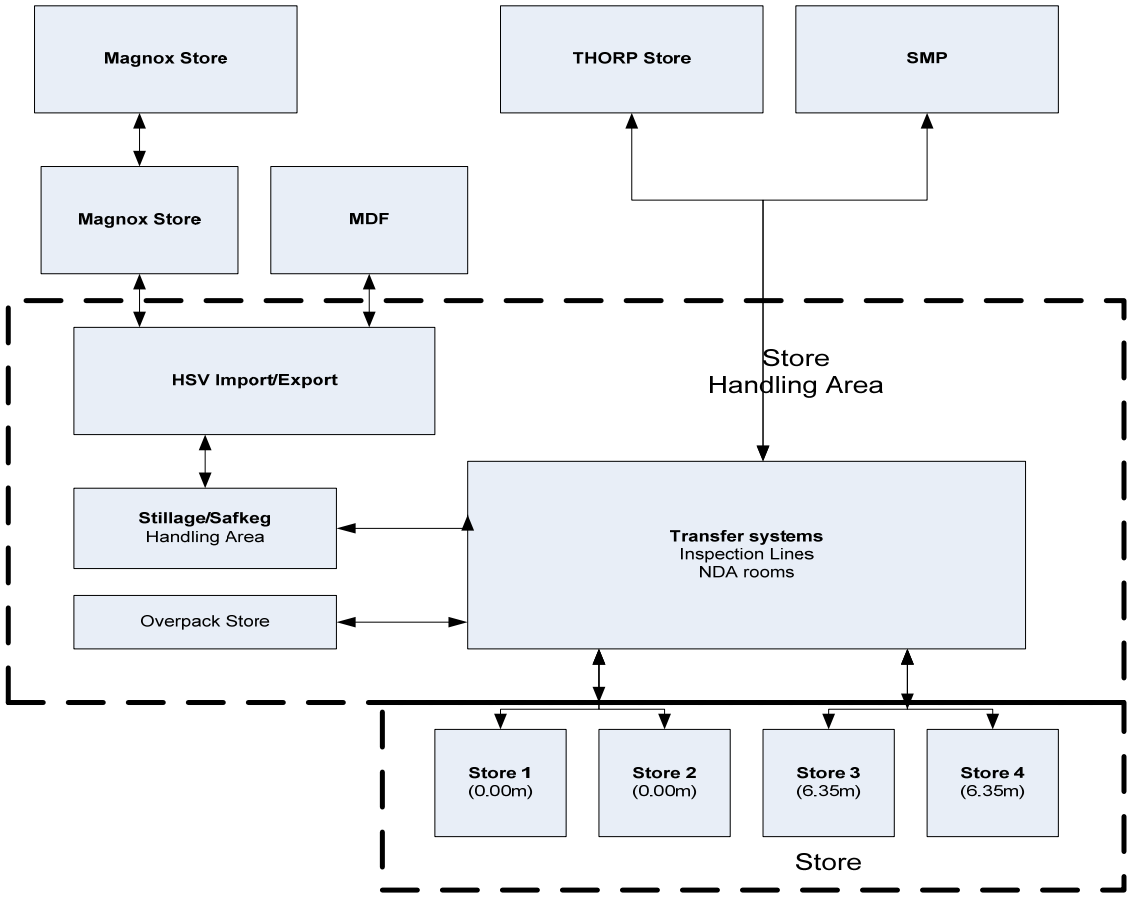

Figure 1: Material flows

4. INSPECTION ACTIVITIES

4.1. Introduction

The inspections will be carried out in accordance with the implementation paper of the Commission reflecting the provisions foreseen in the guidance paper entitled 'A new framework for Euratom Safeguards' discussed at the Working Party on Atomic Questions of the Council of the European Union in December 2005 [1]. These inspection activities would be as follows:

- An annual Physical Inventory Verification (PIV).
- BTC verification during the PIV, including any declared modifications.
- A check of the Nuclear Materials Accountancy & Control (NMAC) records during the PIV.
- 6 – 11 interim inspections, some of which may be unannounced or short-notice inspections.

4.2. Physical Inventory Verification (PIV)

Use will be made of the possibilities for *in situ* verification of cans to provide assurance that cans are present as declared in the channels.

The operator's physical inventory listing will be verified once per calendar year at intervals of not more than 14 months. The following activities will take place:

- Verification of the list of inventoried items (LII).
- The physical verification of nuclear material based upon a statistical sampling plan to meet the detection probability of 60%.
- The use of the installed Cd-Te detector on the transfer trolley to verify *in situ* a number of randomly selected cans that are present as declared in the channels.
- Examination of accounting and operating records, and supporting documents for correctness and self-consistency.
- Establishment of updated book inventory.
- Verification of receipts and shipments.
- Verification of the BTC.
- A review of C/S measures.
- Servicing of surveillance devices if appropriate.
- For the material which has been under C/S up to 5 items may be verified for gross and partial defects.

4.3. Interim Verification activities

A number of interim inspections will be carried out between the annual PIV inspections. These will number between 6 – 11 inspections, some of which may be unannounced or short-notice inspections. The activities will include:

- Examination of accounting and operating records, and supporting documents for correctness and self-consistency :

  (ii) Establishment of updated book inventory once per month.

  (iii) Reconciliation of reports and records.

- Establishment of an updated book inventory after each inspection.
- Reconciliation of reports and records.
- Physical verification of nuclear material based upon a statistical sampling plan to meet the detection probabilities.
- A review of C/S measures including:
  - Application, checking and removal of seals.
  - Servicing of surveillance devices if appropriate.

5. INSTRUMENTATION

5.1. Can Contents Monitor

The Can Contents Monitor is based upon a passive neutron coincidence counter (PNCC) with multiplicity analysis in order to determine the amount of spontaneous fission isotopes (mainly 240Pu and the other even isotopes of plutonium) present in each can. It uses High Resolution Gamma Spectrometry (HRGS) in order to determine the isotopic composition of the plutonium in each can along with the ratio of 235U to plutonium in order that the total plutonium and 235U mass can be determined from the PNCC result. Furthermore it has been designed to allow analysis of all different cans and material types which are expected to be stored in SPRS and is thus quite versatile.

There will be two identical monitors installed in SPRS on the two different floor levels to handle all the material movements into the store and any occasional transfers out.

Within the inspection scheme, the monitors will be used to verify the flow of material into and out of the store material balance area. All cans which enter and leave the stores will be quantitatively analysed. Furthermore the CCMs will be used during the physical inventory verification to re-measure a small number of selected cans randomly selected as required. The units will also provide a potential backup solution in case the containment and surveillance scheme of the stores should ever be compromised

The data acquired at the monitor stations will be collected with the Commission developed data acquisition system (**R**emote **A**cquisition of **D**ata **A**nd **R**eview system). The data will include the neutron coincidence or multiplicity measurements, the gamma spectra and details of the can identification. The analysis of the data will be performed using a specific evaluation package with the acronym CRISP (**C**entral **R**adar **I**nspection **S**upport **P**ackage), a Commission (DG TREN-I) developed data evaluation package. The CRISP software correlates data of different sensors, calculates the measurement results and compares these with the operator declarations. CRISP finally provides a report for the inspector. [2]

5.2. Surveillance scheme – video and neutron monitors

The store part of the facility will be covered by a multiple containment and surveillance system. This will employ a combination of neutron monitors and surveillance cameras that will allow the inspectors to be able to follow the flow of nuclear material into, out and through the store.

Knowledge of the Pu can arrivals into SPRS will commence when they reach the CCM and are subjected to measurement. Their subsequent movement will be monitored and recorded both by digital surveillance cameras and neutron detectors mounted at strategic points inside the storage halls. The high sensitivity neutron monitors to be employed will be similar to those already successfully used in a number of Pu handling facilities within the European Union. Their sensitivity means that they are readily able to recognize movement of items containing nuclear material. The recorded signals can be analysed automatically using a data analysis system and give the inspector a full interpretation regarding the path of movement, which can be compared with the declaration.

As well as the neutron measurement system and neutron monitors an independent video surveillance system (FAST system), consisting of 33 digital cameras, will be installed. The individual cameras have been sited to cover known nuclear material movements as well as possible diversion routes out of the

secure area. The cameras help to provide the containment layer coupled with other monitoring devices such as seals and door closure monitors.

5.3. Electronic Seals

In order to reduce the number of inspection activities and enhance the containment of the different possible access routes seals will be applied as much as possible. The inspector presence in SPRS will probably total only a maximum of 50 days per year so it is important that seals are effectively applied to help maintain knowledge of the store between visits.

One important area of application of electronic seals will be on the emergency exits, which, in principle, should be rarely used. Under normal operating conditions it is not anticipated that the channels will be sealed, nevertheless preparations will be made to cover possible sealing during the initial store loading commissioning phase.

The seal of choice for these applications will be the EOSS seal [3]. This seal can be remotely read out and has been designed with a high level of reliability and security. The readout of the seals will be via the RADAR system and the analysis of the collected data by CRISP.

5.4. Laser Verification

During the discussions at the design and construction phase of SPRS it became clear that due to the size of the ventilation ducting and possible diversion scenario for removal of cans it was important that the containment of the store needed to be checked for its completeness. One particular area of concern was the store cooling ducts that were routed from the 0m level walkways up to the roof level and exhausted into the atmosphere through a series of ventilation stacks. In order to restrict possible access into these stacks the duct outlets were each sealed with a metal grid structure. To ensure that there has been no tampering of this structure the integrity of the upper security grill at the base of each stack will require checking. A Laser Verification technique, developed by JRC Ispra [4] for design re-verification will be applied. This technique based on a 3D scan of the structure will check the layout of the ventilation stack outlet and will confirm whether these grids have been altered or tampered in anyway since the last scan.

5.5. In situ Channel monitor

The cans are placed in storage tubes in the main store on either the 0 m or the 6.35 m level by one of 4 automated charge machines. There is one charge machine for each charge corridor with 4 corridors making up the present SPRS store configuration. In order to provide the inspectors with the assurance that the cans are present as declared in the channels an in-situ verification method has been implemented in the safeguards approach. A number of channels will be selected randomly at the annual PIV to confirm the number and presence of cans as declared within the selected channels.

The in-situ inspection verification of the cans stored within the channels will be by the introduction of a Cd-Te detector that will pass under the row of cans and confirm the presence of nuclear material. The signal from the detector will be interpreted using a standard MiniMCA gamma spectrometer. There will be an in-situ inspection monitor installed on each one of the 4 charge machines.

5.6. Can Identification Verification

The plutonium cans will be identified and tracked using a combination of an Optical Barcode/Character Recognition System (OBCR) system as they enter they enter SPRS through the CCM route. The OBCR system will read either the barcode or the alphanumeric identifications on the cans as they are transferred into and, occasionally, out of SPRS. The system will be a combination of cameras reading the numerical characters on the outer Pu cans and the logic image processing algorithms. The can numbers will be communicated to the SPRS control system via an OPC server.

6. DATA TRANSMISSION

The Regulation (Euratom) N° 302/2005 advocates the use of information technology and of telecommunications networks in the exchange of data between the Commission and operators. The changes to the on-site verification frequency to plants within the European Union since 2005 has re-enforced the need for optimizing the inspectors' work during on-site inspections and the transfer of

part of the verification activities back to headquarters in Luxembourg. The development of this idea implies the need to transfer elements of data to headquarters that were previously only accessible in the installations. Well organized operating records, well structured operators' databases and well defined transfer formats of these data to the DG TREN inspectors are key issues for an efficient use of such data in Luxembourg.

The installed and integrated instrumentation and equipment form a key aspect of the safeguards arrangements within SPRS. Any data transmitted off site back to Luxembourg will need to be structured and targeted for a particular need so that the best use can be made of this information in both the preparation for an inspection and the subsequent inspection and evaluation process.

The on-site inspections activities will be optimised by the use of analysing the transmitted signals of the electronic monitoring, electronic seal status, selected CCTV images signals and the CCM measurement system as well as some other key monitoring devices such as door monitors from the plant back to headquarters.

The use of inspection data transmitted back to headquarters will enable DG TREN to modify the modalities of the inspection verification scheme as the store becomes full and nuclear material transfers into and out of the store become infrequent. This anticipated improvement in efficiency may result in either reduced number of inspection man days as well as the number of visits per annum.

7. CONCLUSIONS

The early interaction between the designers and the safeguards inspectors involved in the SPRS project has helped to facilitate the following features:

- The arrangements for design verification during the construction phase
- The arrangements for design verification during operations.
- Accountancy verification.
- The implementation of containment & surveillance measures.

The proposed inspection scheme using a combination of verification measurements, supplemented by Containment and Surveillance measures, takes into account the safeguards modalities as described in the document 'A New framework for Euratom Safeguards' and discussed at the Council of the European Union in December 2005.

The installed safeguards equipment and components will utilise the present state of the art instrumentation and provide a modular platform for subsequent upgrading during the lifetime of the plant.

The installed safeguards instrumentation will be commissioned in the autumn of this year 2009 with a view to implementation in 2010 when the first nuclear material is introduced into the SPRS plant.

The implementation of the remote data transmission for the SPRS plant will provide valuable assistance in improving the preparation and overall inspection efficiency.

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An Integrated Approach for the Verification of Fresh MOX Fuel Assemblies at LWR-MOX Recycle Reactors

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Abstract:

This paper presents an integrated approach for the verification of mixed oxide (MOX) fuel assemblies prior to their being loaded into the reactor. There is a coupling of the verification approach that starts at the fuel fabrication plant and stops with the transfer of the assemblies into the thermal reactor. The key measurement points are at the output of the fuel fabrication plant, the receipt at the reactor site, and the storage in the water pool as fresh fuel. The IAEA currently has the capability to measure the MOX fuel assemblies at the output of the fuel fabrication plants using a passive neutron coincidence counting systems of the passive neutron collar (PNCL) type [1]. Also, at the MOX reactor pool, the underwater coincidence counter (UWCC) has been developed [2] to measure the MOX assemblies in the water. The UWCC measurement requires that the fuel assembly be lifted about two meters up in the storage rack to avoid interference from the fuel that is stored in the rack. This paper presents a new method to verify the MOX fuel assemblies that are in the storage rack without the necessity of moving the fuel. The detector system is called the Underwater MOX Verification System (UMVS) [3]. The integration and relationship of the three measurements systems is described.

Keywords: MOX; verification; underwater; NDA

1. Introduction

With the increase in the use of light water reactors (LWR) that are fuelled with mixed oxide (MOX) fuel assemblies, it will be necessary to apply effective safeguards and material control for the fuel assemblies. Safeguards measurements are required for fresh MOX fuel assemblies at both the fabrication plants and reactor sites. An important part of the cycle is the transfer of the assemblies from the fabrication facility to the reactor sites, and the subsequent storage of the assemblies underwater in the reactor spent fuel storage pool prior to use. There needs to be effective verification that the plutonium mass remains intact during the transfer and subsequently in the underwater storage location.

This paper describes an integrated safeguards approach for the verification of fresh MOX fuel assemblies for the LWR-MOX fuel cycle based on three Non-Destructive Assay (NDA) systems. At the MOX fabrication facility, the passive neutron collar system is currently used as an in-plant NDA system to measure the finished fuel assemblies. For the MOX assemblies that have been shipped to a reactor site, the underwater coincidence counter (UWCC) has been used by the IAEA for the shipper-receiver verification in the underwater storage. Both of these instruments are based on neutron coincidence counting to verify the plutonium mass. For the interim verification of MOX fuel during water storage, a new NDA system called the underwater MOX verification system (UMVS) is presented in this study. The UMVS is designed to be lowered into the water to rest on the fuel assembly storage rack and verify the plutonium content in a target MOX fuel assembly. The new system is based on singles neutron counting, and it eliminates the necessity of the fuel assembly movement. The neutron cross-talk from neighbouring fuel assemblies is corrected for by the configuration of the multiple ^{3}He neutron

detectors. The design characteristics, expected performance, and applicability of the NDA systems for the MOX fuel fabrication facility and for the fuel storage pool at LWR reactor site are discussed in this paper.

2. Integration concept

Figure 1 shows the relationship between the three measurements systems, and the connection to measurement standards at the MOX fabrication plant. The primary measurement takes place at the MOX fuel fabrication plant where a measurement precision of better than 1% is possible for installed systems such as the Fuel Assembly Assay System (FAAS) at the PFPF (Plutonium Fuel Production Facility) in Japan [4]. The IAEA is able to create an independently verified MOX fuel assembly standard during the fabrication process by sampling pellets and maintaining continuity of knowledge (C-o-K) for the standard fuel assembly. After the fuel assembly transfer to the reactor site, the UWCC can measure the ^{240}Pu effective as well as the neutron multiplication of the MOX assembly in the water. The ^{240}Pu calibration is linear after the multiplication correction as illustrated in Figure 2. The data shown in Figure 2 represents four different types of MOX assemblies with very different Pu loadings. The low mass data corresponds to a mock-up assembly at the SCK-CEN VENUS facility in Mol, Belgium; the mid range data corresponds to a MOX mock-up assembly at LANL; and the high mass data points correspond to data from MOX reactors in Europe.

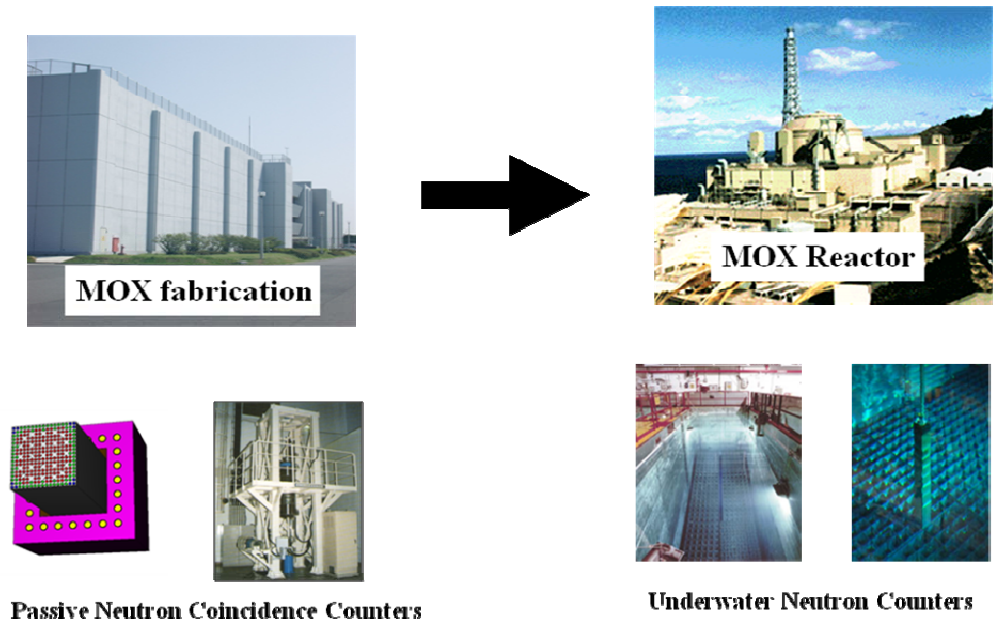

Figure 1: Fresh MOX fuel assembly shipper-receiver verification and storage pool verification.

There are three basic sources of neutrons from the MOX fuel assemblies:
1. the spontaneous fission neutrons from the 240Pu effective
2. the alpha,n reaction source neutrons from the plutonium oxide, and
3. the induced fission multiplication(M) from all sources of neutrons.

Because the UMVS measures the same basic neutron signature as the preceding verification systems (PNCL and UWCC) at the fabrication plant and during fuel receipts, namely the ^{240}Pu effective, the overall verification measurement can be considered as a neutron balancing approach between the fabrication plant, the shipment, and the underwater storage prior to filling the reactor core. This paper will describe the three measurements systems together with their interrelationship in the safeguards verification of the fuel assembly product in the advanced MOX reactor fuel cycle.

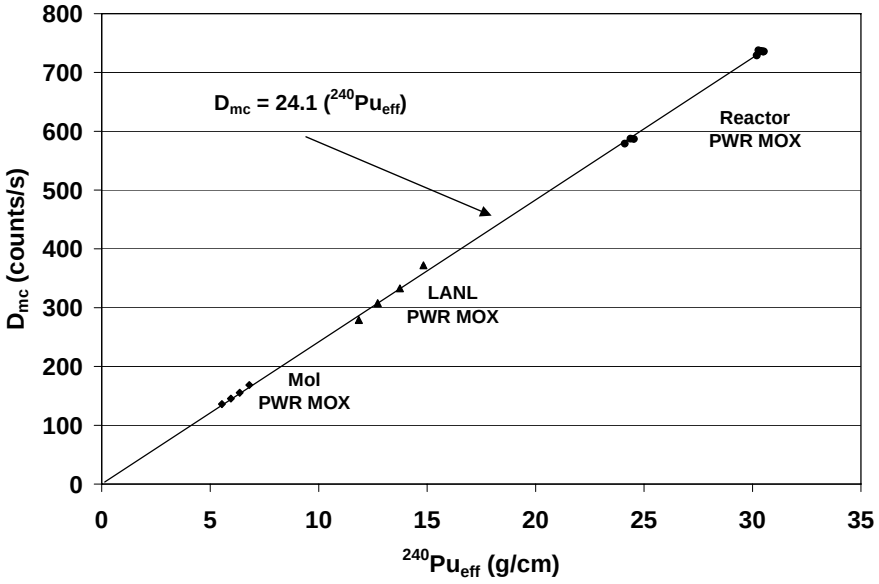

Figure 2: The UWCC calibration curve for the multiplication corrected doubles rate for the verification of the fresh MOX fuel assemblies in underwater storage.

3. Primary Measurement System

The PNCL shown in Figure 3 was developed to measure the ^{240}Pu at the MOX fuel fabrication plants in the attended mode. The instrumentation has been in use by the inspectorates for more than two decades. The neutron singles and doubles counting rates are measured and the neutron multiplication is calculated. More recently, the PNCL has been used by the IAEA in France to measure MOX assemblies prior to shipment providing an accuracy of ~ 1%. A similar measurement system, the FAAS, is installed at the PFPF in Japan (see Figure 4) for the IAEA verification of the Fast Breeder Reactor (FBR) MOX assemblies. This system operates in the unattended mode with continuous data collection. The MOX fabrication plants require a safeguards verification of the plutonium entering and leaving the plant. The measurement of the fabrication plant output is the first step in the integrated approach illustrated in Figure 1.

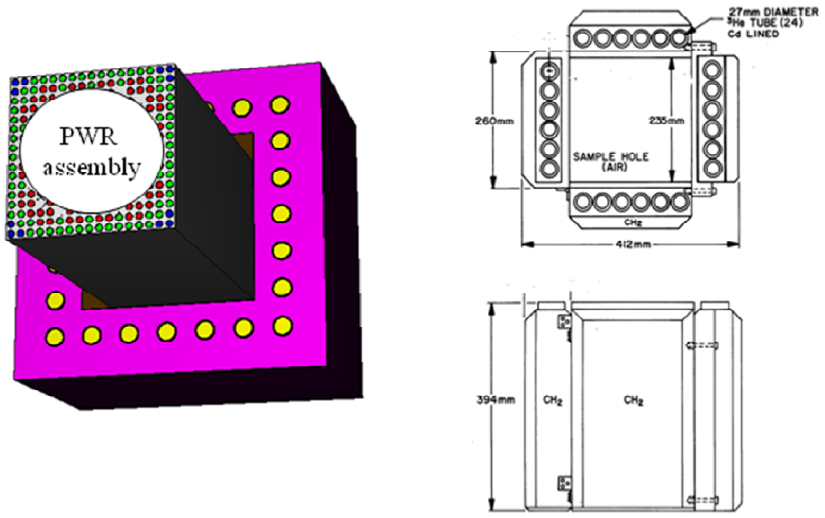

Figure 3: Schematic diagram of the Passive Neutron Coincidence Collar for application to MOX fuel assemblies.

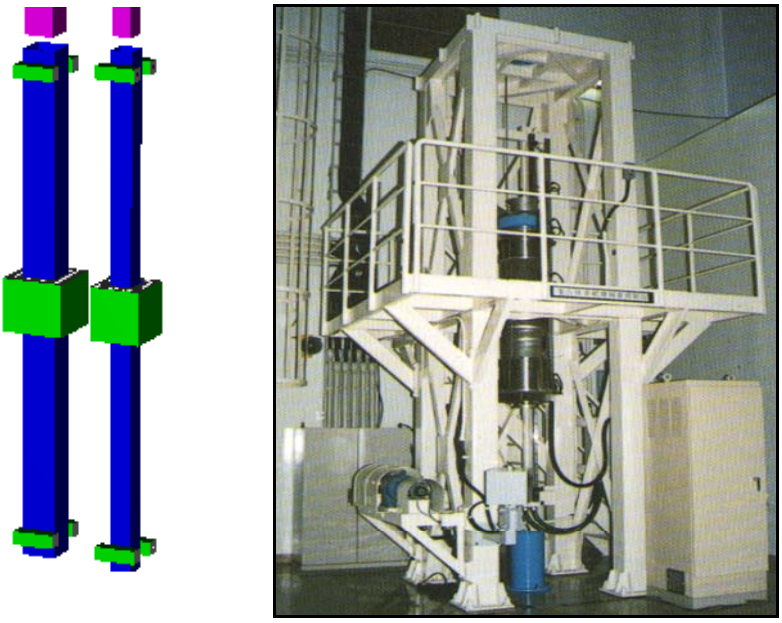

Figure 4: Fuel Assembly Assay System (FAAS) for MOX assembly.

4. Secondary Measurement System

4.1 The UWCC

The transfer of fuel assemblies from the fabrication plant to the reactor site is an important step for safeguards and material accountancy. Sometimes this transfer will cross national boundaries adding to the importance of the verification measurements. The MOX assemblies are placed in the storage pool at the reactor sites for protection and security. The UWCC, shown in Figure 5, has been used by the IAEA to verify the Pu content at the MOX assemblies in the underwater location. The UWCC is packaged similar to the Fork detector [5], except the fission chambers have been replaced by ^{3}He neutron detectors. The UWCC measures the fuel assemblies at positions that are above the storage rack, and after the UWCC measurement, the fuel assembly becomes a working standard for subsequent measurements using the UMVS.

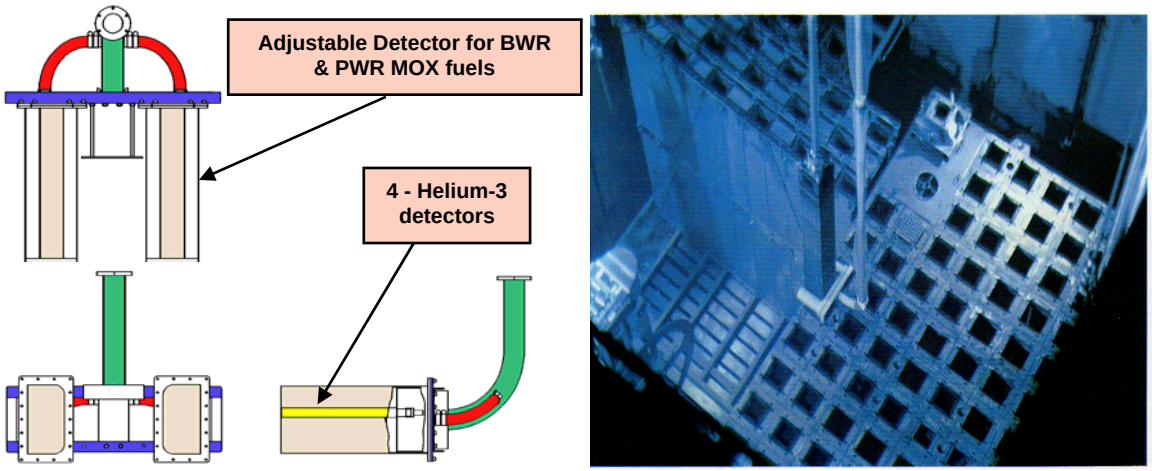

Figure 5: The UWCC detector head for verification of fresh MOX fuel assemblies in water.

4.2 The UMVS

The proposed UMVS was designed to provide a verification of the MOX assemblies in the underwater storage rack without the necessity of moving the fuel. The goal of the UMVS is to verify that the fuel assembly has not been modified after it was placed in the storage rack. The UMVS is not designed to measure the total Pu in the MOX assembly in that the measurement is only sensitive to the Pu in the upper third of the assembly. The capability of the associated UWCC, that is part of the IAEA instrumentation suite, limits the more extreme diversion scenario of modifying only the bottom section of the fuel assembly.

The UMVS measures the total neutron emission from the top of the assembly. The total neutrons that can be measured near the top of the fuel assembly is a combination of the three source terms listed above. The neighbouring assemblies in the storage rack also contribute to the neutron flux. The UMVS presents a method to correct for the cross-talk from the neighbouring assemblies in the rack. The system uses neutron detector ratios to correct for neutrons from the neighbouring assemblies in the storage rack. To obtain a high counting efficiency, the ^{3}He detectors are embedded high density polyethylene (HDPE) moderator with an air collimation channel to the centre detector. Figure 6 illustrates a cross sectional cut through the UMVS showing the central 5-cm diameter ^{3}He tube and the five perimeter ^{3}He tubes with 25-mm diameters. All tubes are 12.5-cm long with a ^{3}He pressure of 6-atm. The purpose of the perimeter tubes is to measure the neutrons from the neighbouring assemblies in the storage rack so that a correction can be made to the counting rate in the central ^{3}He tube. The correction algorithm is developed in the next section.

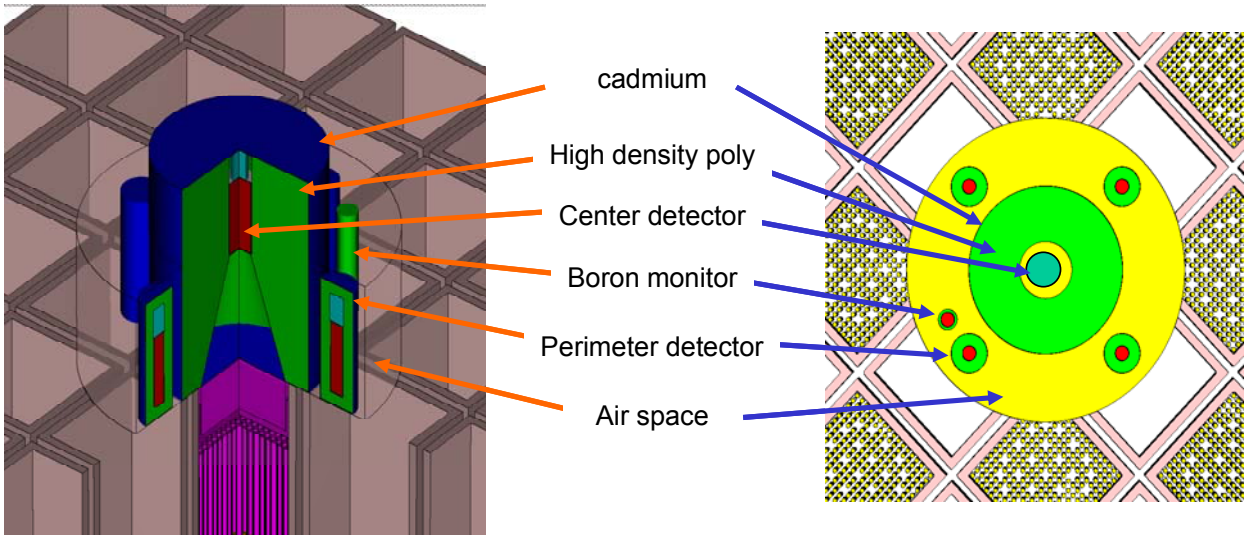

Figure 6: MCNPX depicted the UMVS on the top of a fuel storage rack.

The measured rate of the neutron emission from the fuel assemblies is reduced when boron is added to the water. The water in the storage pool for PWR fuel assemblies typically contains 2200 ppm of dissolved boron. This boron in the water is to reduce the reactivity for the fresh fuel assemblies, and is normally used for the fresh PWR MOX assemblies. The UMVS has an additional ^{3}He detector in the perimeter to verify the declared boron concentration. The extra tube has minimal HDPE moderator so the ratio of the un-moderated and moderated ^{3}He tubes is sensitive to the boron concentration.

4.2.1 Theoretical approach

The singles count rate S from the target assembly would depend on the mass of ^{240}Pu, neutron multiplication, and (α,n) neutrons. The overall multiplication in the storage rack is a function of number of assemblies, the fissile mass, and the storage pattern of the assemblies, and boron concentration of the water.

To develop a theoretical approach for UMVS measurement, following assumptions are made.

1. The neutron multiplication of an isolated target assembly, M_T is a known function of the mass by MCNPX simulation and calibration.
2. The space made by diverted fuel pins in a target assembly is filled with dummy pins, which are filled with depleted uranium oxide of the same density as the original MOX fuel pins.
3. The overall neutron contribution by neighbouring assemblies is considered as an additional neutron source, which is a function of number of assembly and the storage pattern.

With these assumptions, the singles count rate at a target assembly that is surrounded by multiple assemblies can be described by Eq. (1). A new term, β, is introduced to define the overall excess neutron sources including (α,n) neutrons from the target assembly and all neutrons from the neighbouring assemblies. The term β can be measured using the perimeter neutron detector configuration in the UMVS.

$$S' = m_T \cdot f \cdot \nu \cdot \varepsilon_T \cdot M_T \cdot (1+\beta) , \quad (1)$$

where, m_T = mass of ^{240}Pu in target assembly,
f = spontaneous fission yield of ^{240}Pu (=473.5 fission/g-s),
ν = spontaneous fission multiplicity of ^{240}Pu (=2.16),
ε_T = efficiency of the neutron counter for target assembly,
M_T = neutron multiplication for the target assembly,
β= ratio of overall excess neutron to the spontaneous fission neutrons.

With Eq. (1), the singles count rate at a target assembly can be used to verify the content of the target assembly.

4.2.2 MCNPX simulations

The UMVS system illustrated in Figure 6 was simulated using the MCNPX code [6] to guide the design and to estimate the counting rates from the various detectors. The calculations were also used to develop a correction algorithm for the neighbour assemblies in the storage rack.

Because the measurement requires a correction for neutrons originating in the neighbouring assemblies, four side ^{3}He detectors were introduced to measure the effect of the neighbors. The combination of the polyethylene and cadmium sheet is also providing shielding to the side detectors for neutrons that originated from the target assembly.

The side detectors are located at four equally spaced positions around the perimeter of the system and covered with 2-cm thick annulus of polyethylene covered by cadmium (Cd). The perimeter detectors are embedded in the annular space that surrounds the main body of the detector. The distance of the side detectors from system's center detector is optimized for maximum efficiency for measuring neutrons from the neighbor assemblies and minimum efficiency to the target assembly. One of the side detectors is accompanied with an additional detector for boron concentration monitoring. The fifth detector is isolated from the HDPE in the UMVS detector body by cadmium sheet of 0.5 mm thickness so that the detector response is sensitive to the boron concentration in the water. Because the two detectors are positioned at about the same location, the neutron flux at the detectors is approximately the same regardless of orientation of the system, so the boron concentration can be verified by comparing two detectors' responses. Because of the buoyancy force of the water, total net weight of the system in water is approximately 5 kg.

The PWR MOX fuel assembly that was used for the simulation has the same geometrical configuration as the Westinghouse 17×17 PWR fuel assembly design. The assembly is composed of low, middle, and high Pu content fuel rods, and the average Pu fissile content is 11 wt%. To reduce the computing time to get a reasonable error (< 5 %), only the top 100 cm of the fuel is considered in the simulations. Additional calculations were performed to investigate the penetration depth of the neutron signal from the top of the assembly. The plutonium sensitivity decreases as a function of distance from the top of the assembly with limited signal sensitivity down to ~ 0.6 m.

A storage rack of 7×7 assembly array in a storage pool is modeled for this simulation. Borated stainless steel of 1.6 cm thickness is used for the structural material of the storage rack. Potential assembly storage patterns are illustrated in Figure 7. The centre circle in the Figure 7 is the location of the target assembly at which the UMVS is located for the verification measurement.

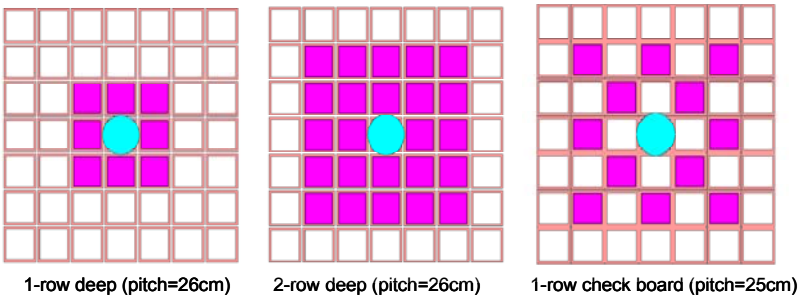

Figure 7: Fuel assembly storage patterns used for the MCNPX simulations.

4.2.3 Results

Figure 8 shows the normalized multiplication of the target assembly as a function of ^{239}Pu mass for 2200-ppm boron concentration. The neutron multiplication and the detector efficiency are for the target assembly that contains 89.9 kg of Pu, which is equivalent to 36.83 kg of effective ^{240}Pu. The multiplication and efficiency of the detection system could be determined by a calibration measurement using one of the MOX assemblies in the pool that was measured using the UWCC. We have used the MCNPX code to calculate the multiplication and detector responses. For the computational experiment, a single assembly in borated water of 2200 ppm is considered as a base case.

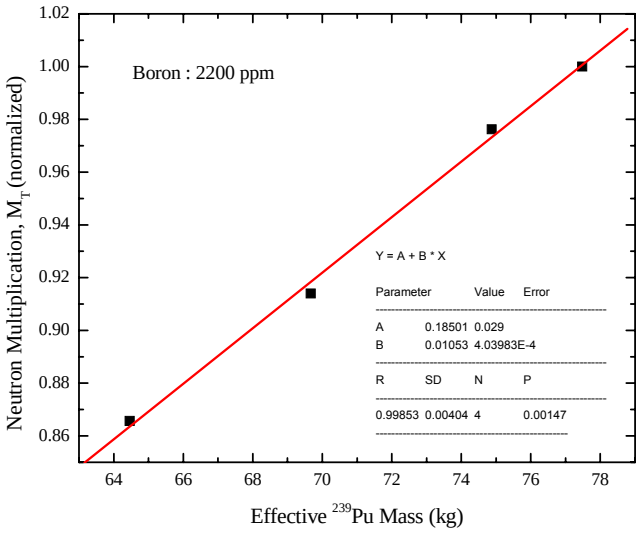

Figure 8: Neutron multiplication by ^{239}Pu mass in a target fuel assembly.

The measured neutron totals rate is a function of (1+β) as shown in Eq. (1), and it needs to be calculated for a given storage pattern and boron concentration. The response of (1+β) can be estimated by the function as shown in Figure 9. We note that the neutron gain by multiplication in checkerboard pattern (see Figure 7) is only slightly greater than the neutron loss by absorption. The

more dense storage pattern (1-row deep) shows the strong neutron contribution from neighbour assemblies, but it eventually becomes saturated at the 2-row deep case because of the separation distance and the absorption of neutrons in the water.

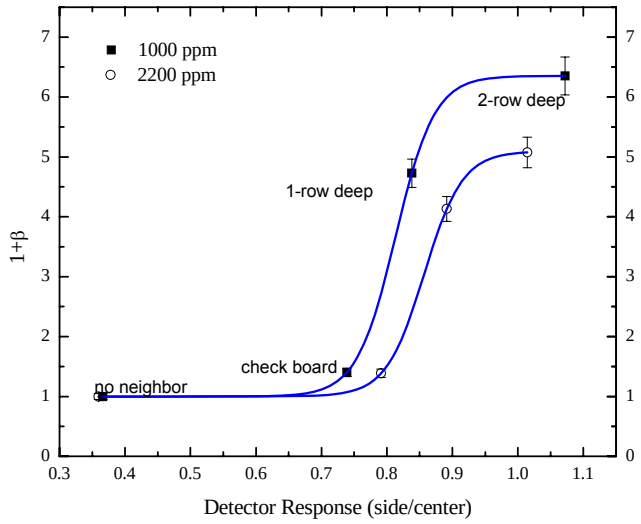

Figure 9: Response of (1+β) as a function of storage pattern.

The boron monitor of the UMVS can verify the concentration by measuring the Cd ratio in the two perimeter detectors as shown in Figure 10.

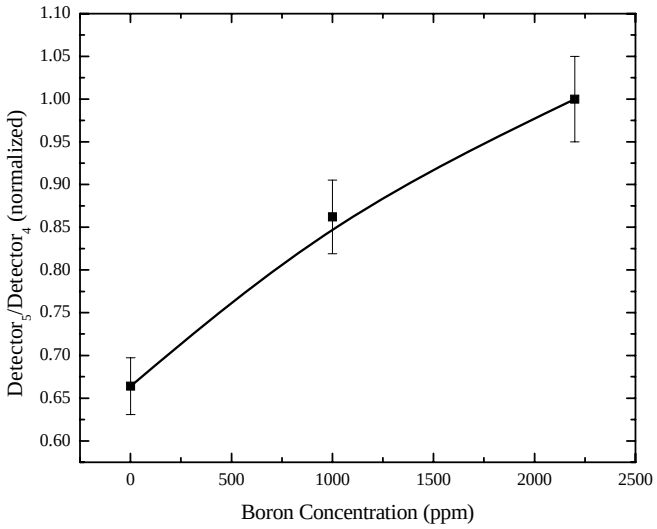

Figure 10: Response of UMVS for boron concentrations.

Figure 11 shows the averaged singles rates after (1+β) correction factor that is based on the response ratio between canter and perimeter detectors. Because the (1+β) is correcting the excess response from neighbour assemblies, the corrected $S_{(1+\beta)}$ become a single response curve regardless of storage patterns. It was determined that ~1 % of random error could be achieved with less than 100 sec of counting time.

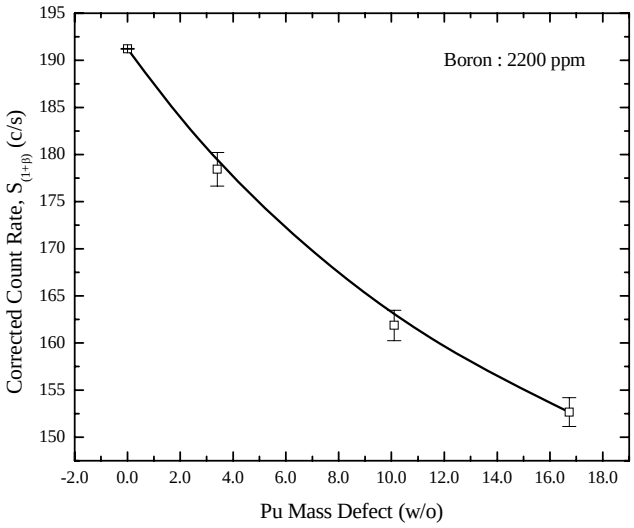

Figure 11: The corrected counting rate from the central ^{3}He tube versus the Pu mass removal from the MOX assembly.

The response of the UMVS has a slight dependence on the radial location of the pin removals and the curve shown in the Figure 11 corresponds to a uniform removal pattern. If the pins are removed from the central section of the assembly, the change in the measurement will be greater than the values shown, and if the pins are removed from the extreme corners, the response change will be less than indicated. However, the measurement is sensitive to all pin positions. In general, the UMVS should be used to verify the consistency of the declared loading and not to quantify the remaining Pu mass.

5. Summary

The verification of MOX fuel assemblies can benefit from the coupling between the fabrication plant measurements and the measurements at the reactor site. These measurements at the fabrication plant and the reactor site are useful to establish effective safeguards for shipper-receiver accountability. In general, NDA measurements make use of one or more standard samples to calibrate the NDA measurements. The proposed connection between the fabrication plant and the reactor site can provide working standards for the subsequent fuel verification using the UMVS.

The PNCL type measurement at the fabrication plant can provide a precision of better than 1% in 10 min. The accuracy will depend on how well the standard is known and its match to the unknown assemblies. The UWCC has a precision and accuracy that is similar to the PNCL. On the other hand, the UMVS is used to measure the fuel assemblies after they are lowered into the storage rack. If C-o-K is maintained between the fuel transfer to the storage rack and the initial UMVS measurement, then the initial measurement provides a calibration for the UMVS. The precision of the UMVS is ~ 1% in 100s; however, there will be additional errors related to positioning the detector head and the correction for the neighbouring fuel assemblies. For the case where there is no fuel movement between subsequent measurements from more than one inspection period using the UMVS, the relative accuracy of the UMVS will be as good as the positioning reproducibility. In this case, the UMVS measurement is similar to a fingerprint type of verification.

For the case of fuel pin substitution, the change in the UMVS response is almost linear with the fuel removal mass as shown in the Figure 11. Note that a 10% fuel pin reduction results in a 10% change in the UMVS response. The sensitivity to the position of the pin is small where the outside perimeter pins have less ^{239}Pu content than the central pins. There are no blind spots in the response. In the vertical direction, the top of the fuel assembly dominates the response with penetration to a depth of about 1 m.

The UMVS has the capability to verify fresh MOX assemblies with a measurement time of only 100 s, and with no movement of the fuel assemblies. The detector package would weigh ~ 40 kg in air and about 5 kg in water, so that its weight on the storage rack is negligible. The detector head could be connected to the bridge crane by a flexible cable for hand positioning.

The status of the UMVS development is that it exists only in MCNPX simulation space, and no prototype hardware has been fabricated or tested. The electronics required to support the UMVS are the same as for the attended mode UWCC (i.e. the JSR-12 or equivalent) [7]. Additional calculations are in progress to better define the spatial response of the system to different fuel storage configurations. As the use of MOX recycle fuel becomes more widespread, the effective and efficient safeguards of the fuel will become more important. The capability to verify the fresh MOX fuel assemblies in the under water storage without fuel movement will take on added importance.

6. Acknowledgements

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Howard O. Menlove

Feasibility study of a neutron monitoring system for a dry storage of RBMK spent fuel

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Abstract:

A dry storage for RBMK spent fuel is under construction at the Ignalina site in Lithuania. Spent fuel will be stored in a special model of casks, called CONSTOR, similar to the CASTOR used for PWR spent fuel, but having a concrete shield instead of cast iron.
This paper will report on a feasibility study performed in order to assess the possibility of monitoring the movements of casks in the dry storage (and in particular entry/exit with discrimination of empty/full casks) using conventional slab neutron monitors.
The paper will present a preliminary assessment based on Monte Carlo calculations and some measurements done on real casks in order to validate the calculations.

Keywords: neutron counting, Monte Carlo, spent fuel dry storage

1. Introduction

A dry storage for RBMK spent fuel is under construction at the Ignalina site in Lithuania. In order to plan and implement Euratom safeguards at the dry storage, the logistical support unit of the TREN-I nuclear inspectorate has commissioned a feasibility study to JRC in order to evaluate the possibility to monitor the movements of casks in the facility. In particular the main goal is to predict whether it will be possible to discriminate empty from loaded casks passing through the storage doors using passive neutron monitors (slab-type).

Spent fuel will be stored in a special model of casks, called CONSTOR, similar to the CASTOR used for German LWR spent fuel, but having a concrete shield instead of cast iron. Some fuel has also been stored in cast iron casks similar to CASTOR, but without the neutron shield and cooling fins. DG TREN has considerable experience monitoring the flow of LWR-loaded CASTOR containers. However, as RBMK fuel has quite different typical enrichment and burnup and will be stored in a different container type, a careful assessment of the viability of the planned monitoring regime is required.

RBMK fuel bundles are approximately 7-m long, so they would require huge casks to be stored. Generally RBMK assemblies are composed by two active sections connected through a central rod (see figure 1). In order to ease their storage, the assemblies are cut in two halves, each approximately 3.5-m long.

In order to predict the neutron fluence outside a CONSTOR cask filled with spent RBMK fuel a Monte Carlo model of the cask, fuel and detectors has been developed as described in section 2. The results have been validated by comparison with a campaign of measurements described in section 3.

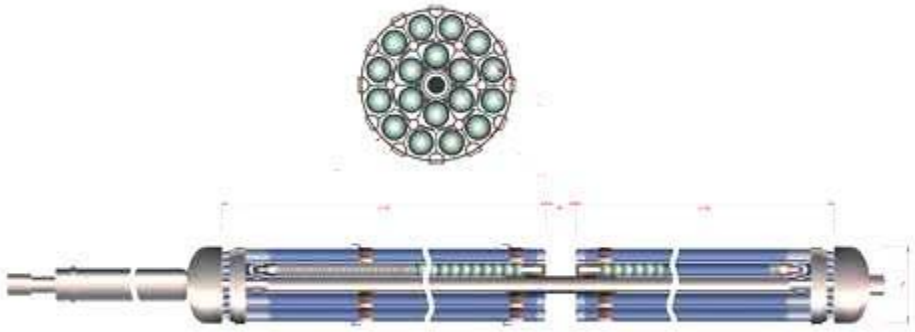


Figure 1 – RBMK fuel element

2. Monte Carlo model of the cask/fuel/detector

The Monte Carlo model of the CONSTOR container has been developed according to the description in the GNS report “CONSTOR RBMK1500/M2 – Ignalina NPP – Shielding Analysis”, GNB B 110/2006 [1].

Three different models with increasing detail have been built:

- a homogeneous (“HOM”) model where the entire fuel and the baskets have been lumped within an homogenized basket region (figure 2);
- a simple heterogeneous model (“HET1”) where the basket is fully modelled, but each fuel bundle has been homogenized within its support tube (figure 3);
- a double heterogeneous model (“HET2”) where each fuel bundle has been described in detail with all individual fuel rods (figures 4 and 5).

The HOM and HET1 approximated models overestimate the neutron flux outside the container by nearly 20% and 10% respectively. Nevertheless they proved to be useful because they can be run in a shorter time: the HET1 model requires counting times more than double with respect to HOM, and HET2 is four times more time consuming. So the simplified models have been used for quick parametric estimations and trend analysis, whereas reference calculations have been done with the detailed HET2 model.

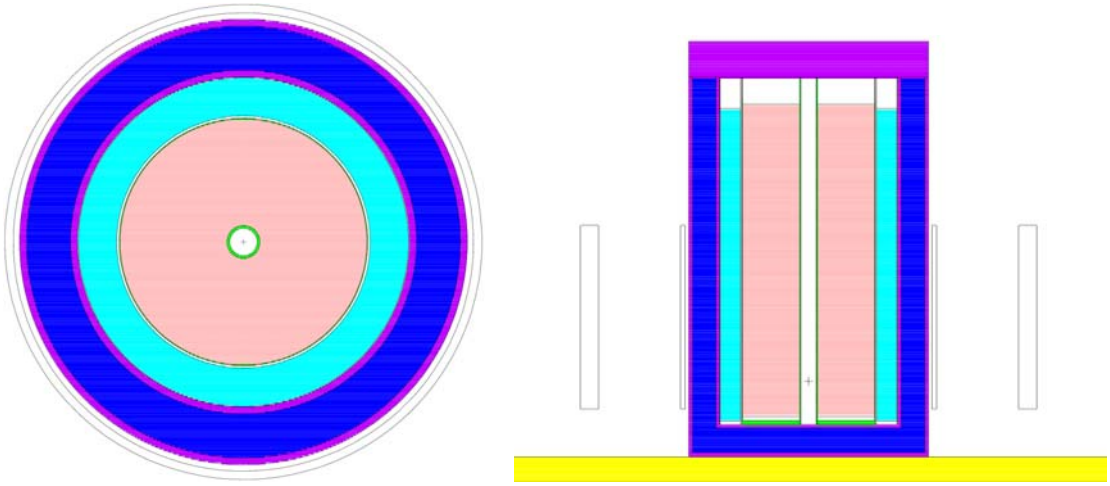


Figure 2 – Model of the homogenized CONSTOR (HOM)

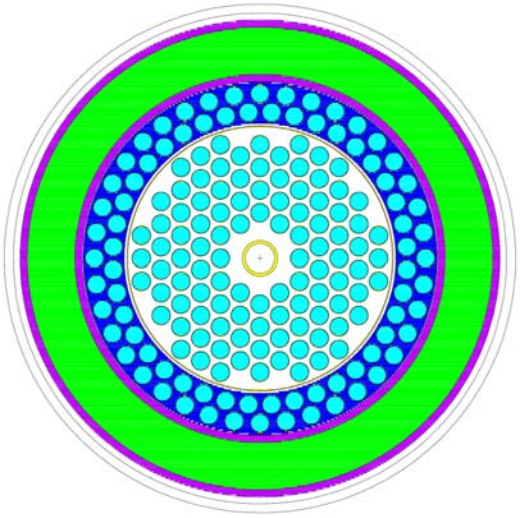


Figure 3 – HET1 model

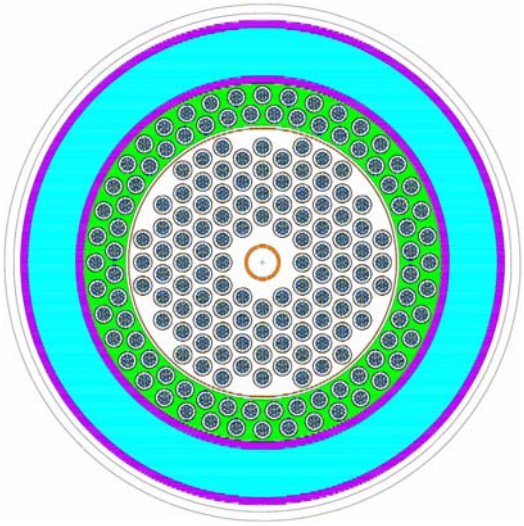


Figure 4 – HET2 model

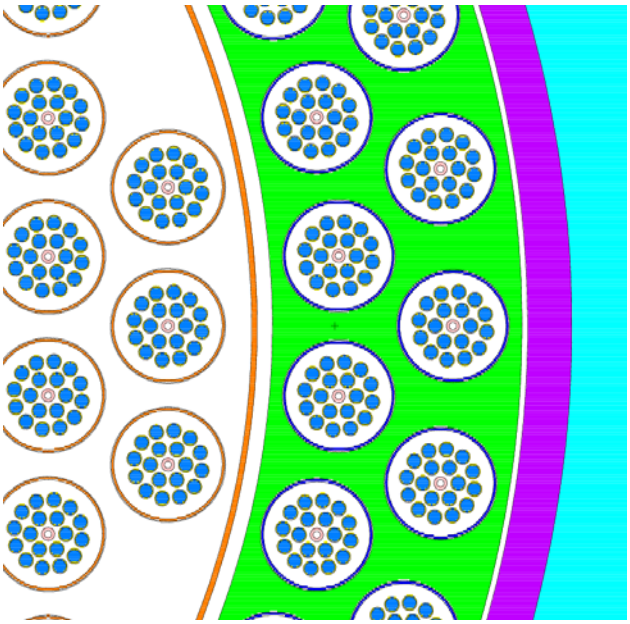


Figure 5 – Detail of the HET2 model

3. The source term

Reference [1] provides a neutron and gamma source term for CONSTOR containers under two different loading schemes:

- LSA with 32M basket filled with 102 (2%-enriched) bundles and ring basket filled with 80 bundles of any enrichment (up to 2.8%)
- LSB with up to 170 bundles (90 in 32M and 80 in ring basket) with enrichment 2%, 2.4% or 2.6%

For the two schemes the total neutron source is quoted at 2.87E+09 n/s for LSA and 3.87E+09 n/s for LSB.

SCALE calculations have been performed in order to check these values, starting from the known geometry and composition of fresh fuel and from the average burnup values provided by the NPP operator. These are reported in the first part of table 1. Pay attention to the fact that each RBMK fuel assembly is composed by two 3.5-meter long active bundles. CONSTOR can host only bundles, so fuel assemblies must be cut in two pieces and each CONSTOR position hosts a bundle (equivalent to half FA). Burnup values reported in table 1 refer to a full fuel assembly (FA).

The results from SCALE calculations in average conditions resulted in a much lower neutron source than the one reported in [1]. This is not surprising since the goal of report [1] was to perform a shielding analysis and dose calculation, so it is quite obvious that conservative assumptions were made in maximizing the source term (probably corresponding to the neutron source from fuel with maximum burnup and zero cooling time). Indeed the source term of reference [1] has been reproduced with SCALE calculations assuming higher burnup values than those declared by the operator, see last row in table 1. Table 2 report the source term calculated with SCALE for “conservative” loading schemes (maximum burnup and zero cooling time) and for realistic conditions (average burnup and 5 years cooling time). The maximum values are consistent with reference [1]. The minimum corresponds to a loading with 2.0%-enriched FA’s only an 10 years cooling time.

Burnup (MWd/FA) / Enrichment	2.0%	2.4%	2.6%	2.8%
Average (declared by NPP operator)	1700	2400	2650	Not av.
Maximum (assumed)	2100	Not used	3500	3800

Table 1 – RBMK fuel burnup used in SCALE calculations

Loading scheme	LSA	LSB
Maximum	2.83E+09	3.86E+09
Average	2.51E+08	2.89E+08
Minimum	6.38E+07	-

Table 2 – Total neutron source in a full CONSTOR

Nevertheless we need to underline the fact that what is conservative for dose calculations is not necessarily justified for designing a monitoring system. In our case realistic conditions would be more suitable and in case we would like to be conservative, probably we should select what is the worst case for a monitoring system: the minimum source. In order not to proliferate with too many scenarios, all calculations have been performed using the maximum value from reference [1].

Due to the linearity of the response with the source term, all the results reported can be scaled down to any “true source” by simple proportionality. It is therefore enough to keep in mind that realistic field conditions for the neutron flux are likely to be a factor 10 to 50 lower than the assumed maximum value used in the simulations.

4. Results from Monte Carlo calculations

Using the three models described above we have computed the expected neutron flux at the external surface of the CONSTOR and at distances of 1, 2, 5 and 10 meters. These are reported in table 3.

Model / distance	contact	1 m	2 m	5 m	10 m
HOM	338	131	74	22	6.9
HET1	314	121	66	20	6.3
HET2	280	107	60	17	5.6

Table 3 – MCNP-computed neutron fluxes (n/cm^{2}s) outside a CONSTOR full container

Considering that a N50 neutron monitor has an effective area of 750 cm^2 (approximately 25 cm x 30 cm), several thousand of neutrons per second will reach the monitor even at a distance of 10 meters, enough to give a clear signal considering that the efficiency of N50 varies between 1% and 10% depending on the energy of the neutrons.

MCNP calculations could in principle be used to predict the count rate in a neutron monitor outside the container. Unfortunately, due to the small size of the detector with respect to the container, the probability of a neutron generated inside the CONSTOR to reach an N50 is of the order of 10^{-7} to 10^{-9}. In order to have a reasonable statistic, we should simulate not less than 10^{+12} neutron histories that would require years of computing time.

In order to reduce the counting time we modelled a huge pseudo-N50 detector. This detector has the same characteristics of an N50 (10-cm thick polyethylene slab with 1-inch diameter 3He tubes at a distance of 5 cm center-to-center from each other) but much larger total dimensions (hundreds of tubes instead of 4 with a length of several meters instead of 30 cm). This trick increases the effective detection volume, but keeping constant the detection probability per volume unit. Therefore the count rate in a real N50 can be simply estimated by multiplying the count rate in the pseudo-detector by the ratio of the volumes of the neutron tubes. Under these conditions the expected count rates in a neutron monitor at difference distances from the CONSTOR have been obtained with good statistics in few hours of computing time. The results are reported in table 4. The “reference” value is referred to the results using the “conservative” value of 3.87E+09 n/s. The minimum and maximum values refer to the range of results expected in realistic cases, obtained by reducing the source term of a factor 13 (corresponding to a cask filled with 2.6% enriched fuel at nominal burnup and 5 years cooling) and 50 (cask filled with 2.0% enriched fuel at nominal burnup and 10 years cooling) respectively.

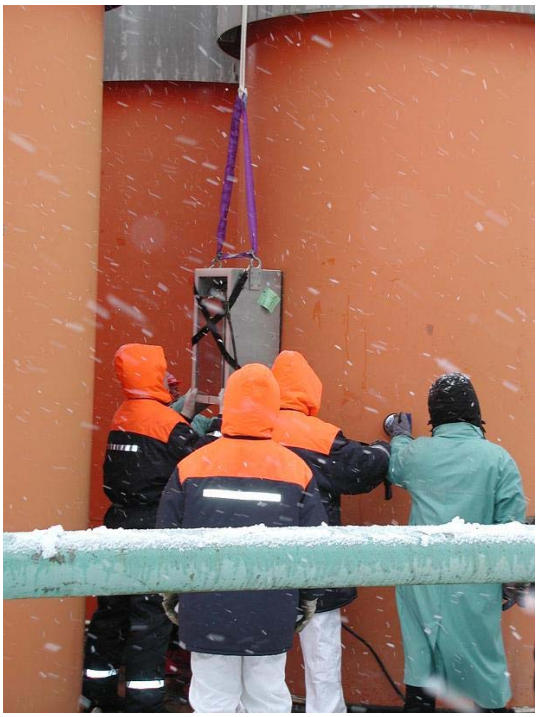
Case / distance	contact	1 m	2 m	5 m	10 m
Reference	20000	4000	3200	600	150
Maximum	1500	300	250	50	12
Minimum	400	80	60	12	3

Table 4 – MCNP-estimated count rates (s^{-1}) in a N50 detector at various distances from a CONSTOR full container

5. Validation measurements

A campaign of measurements on casks stored at Ignalina NPP has been done by TREN inspectors in November 2008.
The spent fuel stored on-site in Ignalina is currently stored in two types of casks:
- stainless steel casks, derived by the CASTOR type used for BWR, but with different design
- CONSTOR casks, but with a design slightly different from the model that will be deployed at the dry storage facility (on which we based our calculations)

The measurements were performed using dual detectors that allow to correct for the strong background caused by the close neighbour casks and were taken nearly at contact with the external surface of the casks (see picture 6).



The (background-corrected) count rates measured at the surface of CONSTOR casks ranged between 400 and 800 counts per second. The rates thus fall completely within the expected range estimated in the previous section. These measurements can be considered a good validation of the estimations done in this paper.

The measurements on the CASTOR-similar casks were quite surprising because they gave results in the range of 17000 to 20000 counts per second, much larger than the one from CONSTOR. This can be possibly explained by the modified design of these casks with respect to ordinary CASTOR's. For instance the cast iron casks in Ignalina have no cooling fins and possibly are not provided with the polyethylene rods for neutron shielding.

6. Conclusions

The values reported in table 4 seem to point to the fact that even at a few meters a full container should give a clearly detectable signal. The access door to the dry storage is relatively narrow (3.5 m), so a neutron monitor strategically placed at the door side could be at an approximate distance of 1 m or even less at the passage of the cask through the door.

Nevertheless the capability of the monitor to see the passage of a CONSTOR will depend on the background, because the discrimination empty/full casks will rely on the increase of count rate in the monitor (signal-to-noise ratio). In the most pessimistic cases (CONSTOR filled with low burnup fuel and very long cooling time), the detection could become problematic in a high background environment.

From the storage plan it results that there are potential storage positions only at 5 meters from the entry door, where presumably the monitor will be installed. This means that if full containers would be stored in these positions, the background signal on the monitor could easily reach the order of 10^2 counts per second. The passage of a container with low burnup fuel and long cooling time would increase the signal of the same order of magnitude and possibly masked by background fluctuations. If these conditions will be confirmed by field measurements, one of the two following options should be taken in order to preserve the efficiency of the monitoring system:

- shielding the N50 from the side of the storage,
- delimiting a "respect zone" surrounding the monitor where no full containers are allowed to be stored.

In conclusion, even though some fine tuning could be needed after real operating conditions will be better defined, an effective monitoring of the movements of CONSTOR casks in the Lithuanian spent fuel dry storage using neutron slab detectors appears realistically feasible.

SESSION 20

KNOWLEDGE MANAGEMENT + TRANSFER

Safeguards Knowledge Transfer: Strategies for Today and Tomorrow

Presented to the 31st Annual Meeting
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Vilnius, Lithuania
May 24-26, 2009

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Abstract:

The Canadian Nuclear Safety Commission (CNSC) and the International Atomic Energy Agency (IAEA) have collaborated on knowledge capture and transfer initiatives during the past several years. Since the late 1990s, both organizations have faced similar challenges arising from the so-called demographic crunch – the sudden retirement of many experienced employees of the "baby-boom" generation. So, both organizations have been interested in finding ways to preserve and pass on their knowledge assets.

A key practical issue for safeguards organizations is the need to capture tacit (personal) knowledge from experienced safeguards employees about to retire or move to other jobs. This paper will offer some recent examples and techniques of successful end-of-career knowledge capture and draw best practices from those experiences.

However, it has been recognized that capturing knowledge from individuals on a crash basis at the very end of their careers is not an efficient and effective strategy. A better approach is to reduce this need by capturing and transferring information on an ongoing basis, which is the essence of good knowledge management.

In support of the IAEA Department of Safeguards, the CNSC undertook a project to introduce the Department's Quality Management System to employees by means of a computer-based training program (QMS-CBT). It became clear in the course of that work that the main ISO standard (9001:2000) chosen by the Department of Safeguards as its framework for quality management lends itself very well to the on-going transfer of both tacit and explicit knowledge assets. Eventually, full implementation of the QMS may obviate the need for "emergency" end-of-career knowledge capture exercises.

Looking to the future, this paper will also explore how an ISO 9001:2000 work environment facilitates the on-going transfer of tacit and explicit knowledge assets in support of individual and organizational learning, decision-making, continuous improvement, and innovation.

Keywords: knowledge management; tacit knowledge; quality management; QMS; ISO 9001: 2000

1. CNSC / IAEA Knowledge Management Initiatives

The Canadian Nuclear Safety Commission (CNSC) and the International Atomic Energy Agency (IAEA) have collaborated on knowledge capture and transfer initiatives for more than a decade. Since the late 1990s, both organizations have faced similar challenges arising from the so-called demographic crunch – the accelerating retirement of experienced employees of the baby-boom generation. So, both organizations have been interested in finding new and better ways to preserve and pass on their job-related knowledge.

The first cooperative CNSC/IAEA initiative in this area was undertaken in 1998 when multimedia technology was used to develop self-paced training materials for field inspectors to reduce the requirement for classroom presentations by senior staff members already facing many other demands on their time.

In the years since, the CNSC has undertaken a number of additional interactive training programs for the Agency and has investigated and applied a variety of knowledge management approaches. These approaches have included video capture of presentations by experts, E-Doing portals, the use of Wikis as a knowledge sharing tool, Communities of Practice, technology-assisted mentoring, and the use of animation to explain processes and abstract concepts, to name a few.

The CNSC has also made an effort to step back from specific techniques to consider big-picture topics. For instance, the CNSC conducted a wide-ranging survey within the Canadian federal government to identify knowledge management programs, resources, and success stories as a means of identifying best practices that can be shared with others. Other work has focused on top-down and bottom-up approaches to implementing knowledge management, developing effective knowledge transfer strategies, and striking a balance between traditional and technology-assisted knowledge transfer methods.

2. Eleventh Hour Knowledge Retention

A key practical issue currently being faced by CNSC, IAEA and other organizations in the nuclear industry is the need to capture information from experienced safeguards employees before they retire or move to other jobs. This is happening in unusually large numbers right now due to demographic factors. This transition period comes at a time when new protocols and the increasing complexity of systems and methodologies employed by the nuclear industry require that staff at all levels have ever-greater knowledge and skills. Together, these two trends pose significant challenges for the retention and transfer of essential knowledge.

Where does this essential knowledge reside? Explicit, codified knowledge is resident in written policies, procedural manuals, debriefing reports, and a host of other documents and information repositories in both paper and digital form. Implicit or tacit knowledge – also called experiential knowledge – is largely resident in the heads of experienced employees. It is the second kind of knowledge -- tacit knowledge -- that poses particular challenges for knowledge retention.

When a particularly valued individual is about to retire, the method of knowledge retention that often comes first to mind is an interview – or perhaps a series of interviews – to capture the experiential knowledge that will otherwise depart the organization along with the individual. However, experience in trying to capture information at the “eleventh hour” in this way has generally proven to be ineffective largely due to the lack of a systematic process for ensuring that valuable information is, indeed, captured. In the absence of such a process, the individual is often at a loss to know what to say – and interviewers frequently do not know what to ask. Further, once the interviews have been captured, organizations may be at a loss to know how to make this knowledge available to other staff in a convenient, compelling way.

In an effort to address the related issues of capturing useful information and of making it readily available, the CNSC has been experimenting for some time with video-based interviews structured for delivery as short, self-paced tutorials via the Web. Our first such experiment was in 1998, when the Commission

hired a professional video crew to tape a series of 11 short presentations on signal interpretation – none more than five minutes in length – by the senior engineer responsible for development of the VIFM Integrated Fuel Monitor used in CANDU power plants to monitor radiation discharges. The topics were discussed and roughly scripted with the engineer in advance of the recording session. The presentations were then packaged in a CD-ROM-based interactive program that enabled inspectors to selectively view those parts of the presentation of greatest interest to them. The package included a verbatim written transcript of what had been said to enable keyword searches.

The program was well received and widely used. A similar approach has since been used to capture expert knowledge on other topics including CANDU fundamentals, ARC Seal functions and procedures, and, most recently, inspection tips and techniques from a senior international inspector. Additional video-based knowledge capture projects are planned for later this year.

As successful as these individual efforts have been, what has been lacking is a documented, systematic approach that can be widely used with predictable results.

An important step in that direction has recently been taken by the Knowledge Management Coordinating Team within the IAEA Department of Safeguards. The team is in the process of developing and applying a procedure for the transfer of job-related knowledge from departing staff members which:

- defines roles and responsibilities;
- specifies planning procedures;
- provides a method for determining which knowledge to transfer (i.e. who to interview);
- identifies a variety of knowledge-capture techniques appropriate to different circumstances; and
- provides tools such as detailed questionnaires to assist in knowledge capture.

Drawing on emerging best practices in the field of interview-based knowledge capture, the procedure requires that the target individual's manager and colleagues be consulted on what knowledge they would most like to be preserved. This is a departure from past approaches in which the expert him or herself is asked to determine what is important. This often leads to the laborious capture of all sorts of information that is of little instrumental value to successors or clients.

Inviting colleagues to provide feedback of this sort helps to create a map of the retiree's essential knowledge – as perceived by people who will need to use that knowledge after he or she departs.

The procedure is still in its early days of development and application. Several interviews arising from the application of these procedures have recently been conducted and the results are being evaluated in order to further refine the procedures, if required. The CNSC will complement the IAEA's initiative by testing a modified version of the procedure in Canada. The outcomes of these initiatives could be of tremendous value to all of us in this industry.

3. KM in an ISO Environment

Discussion about techniques for 11th hour knowledge capture beg an important question: is this the best way of doing things? Clearly, there are good reasons for approaching knowledge capture in this way in the short run. However, most professionals in the knowledge management business will tell you that a more effective long-term solution is to embed knowledge sharing in day-to-day work retains in a way that reduces, or perhaps eliminates, the need for last-minute efforts to preserve knowledge about to walk out the door.

In 2007-08, a team under the guidance of a CNSC Project Manager worked with the IAEA Department of Safeguards to introduce the new departmental Quality Management System to departmental staff by means of an online training program.

It became clear in the course of that work that the ISO standards that the Department of Safeguards has chosen as its frameworks for quality management have many important implications for knowledge management.

Senior Department of Safeguards management has adopted a Quality Policy Statement, which includes reference to knowledge management as an enabling activity. Quality management has also been endorsed in the Agency's Medium Term Strategy 2006 - 2011. This gives quality management – and by extension, knowledge management – a profile that is lacking in many other organizations. Adoption of the ISO 9001:2000 Standard also means that the Agency is already committed to a management framework that places heavy emphasis on knowledge capture and transfer.

ISO 9000 has become the most commonly-used quality management standard in the world. The ISO 9001:2000 Standard deals with requirements for Quality Management Systems. The 9004:2000 Standard presents Guidelines for Performance Improvements in Quality Management Systems.

The more important of the two from a knowledge management perspective is the 9001:2000 standard because it says that an organization wishing to meet general quality management requirements must develop processes for handling both tacit and explicit knowledge assets. Both are required to support decision-making, continuous improvement, and innovation.

The ISO quality management approach is built on understanding and fulfilling the requirements of internal and external "customers". Each business process has its customers. In this sense, the person who has to make use of a document that I create is my customer. I have to talk to that customer, understand his or her needs, and respond appropriately if I'm going to deliver a product that has quality in that customer's eyes. I, in turn, am a customer for the person providing me with, for instance, research findings that I will use in my document.

Focusing on customer needs and satisfaction is an incentive for dialogue, measurement, continuous improvement, and learning. It is a powerful knowledge capture and transfer process in its own right.

Further, the ISO 9001:2000 framework encourages the creation of a device called a "quality manual", which is a repository for process knowledge. Whether it's a hardcopy document or an online resource, a quality manual facilitates knowledge storage, transfer and application. In effect, the quality manual is a knowledge management database which provides user-centred support for trouble-shooting, decision making and much more.

The ISO 9001: 2000 framework also requires an ongoing collegial assessment process called an "audit", which periodically involves the auditor and process participants in a detailed review of their processes. It is a first-rate means of creating the shared understandings and experiences that are at the heart of organizational memory.

These collegial audits – in which there are no losers and no wish to punish – are a particularly good way of identifying process inefficiencies that can then be corrected. Issues and corrective actions are documented, which helps to preserve knowledge for future workers and forms a basis for on-going dialogue.

Knowledge management – capturing, organizing, and sharing knowledge within an organization – is at the heart of ISO quality management processes.

3. KM Guidelines

As a high-level quality management document, the ISO 9001:2000 standard does not recommend specific tools and techniques. It leaves the selection of an appropriate implementation roadmap to individual organizations to work out within their own context.

Which raises the question: are more detailed guidelines available from other sources? Fortunately, the answer is yes. There are some truly excellent guides available, three of which are described below.

CEN:
The *European Guides to Good Practice in Knowledge Management*, published by the European Committee for Standardization in 2004, provide a synthesis of good knowledge management practices from the private and public sectors and from academia. The Guides avoid the mistake of suggesting that there is only one way to implement knowledge management. Rather, they propose frameworks and approaches, share case studies and success stories, and describe a wide variety of techniques that may be helpful in appropriate circumstances.

The Guides comprise five sub-documents.

Part 1: Knowledge Management Framework, sets the overall context for knowledge management and presents a framework consisting of business processes, core knowledge activities and "enablers".

Part 2: Organizational Culture, explains how to create the right cultural environment for introducing and supporting knowledge management activities. This is a critical piece of the implementation challenge, because asking people to do things differently typically triggers all sorts of defensive responses.

Part 3: Implementing knowledge management provides a methodology to help organizations get started in knowledge management. This is really the heart of the guideline, because it walks the reader through a five-phase knowledge management implementation process: setting up the project; assessing the current state of knowledge assets and flows; designing core elements of the knowledge management solution; implementing the solution; and evaluating and sustaining the knowledge management environment.

Part 4: Guidelines for Measuring Knowledge Management, provides practical advice on how to assess progress in implementing knowledge management practices.

Part 5: Knowledge Management Terminology, explains key knowledge management terms and concepts.

Gleaning some key concepts from these Guides:

- The organization needs to define its mission, vision, and strategy with regard to knowledge management;
- A culture of motivation, in which people are respected, feel a sense of trust, belonging, and empowerment is necessary;
- Knowledge activities must be seen as an integral part of a wider business process;
- Roles and responsibilities must be made clear;
- Individuals need to be acknowledged and rewarded for their contributions; and
- The environment must be conducive for people meeting, working together, and sharing ideas and experiences.

With practices such as these in place, knowledge exchange takes place on an on-going basis. End of career interviews are much less likely to be required in an environment of this kind.

The Australian Standard:
The Australian knowledge management standard was published in 2005. It is entitled Knowledge Management – a Guide.

Like the CEN Guide, the Australian standard was developed by a working group consisting of representatives from the public and private sectors and academia.

It provides an easy-to-read, non-prescriptive guide to knowledge management, which includes a flexible framework for designing, planning, implementing, and assessing knowledge management initiatives. The language is down to earth and very accessible.

A key concept used throughout the document is "the continuum of the knowledge ecosystem" – which is helpful in conveying the idea that knowledge and knowledge management is part of our everyday working environments, not something apart from it.

There are chapters on:

- Mapping an organization's knowledge ecosystem;
- Translating mapping into priorities for action;
- Beginning the implementation process;
- Choosing from among the many enablers available to knowledge managers; and
- Measuring and evaluating the effectiveness of knowledge management.

There is also a thoughtful chapter on anticipating changes in the knowledge-based economy.

The British Standard: PAS 2001 Knowledge Management. A Guide to Good Practice.
Another useful resource is the British Standards Institute knowledge management standard, entitled *Knowledge Management: A Guide to Good Practice*.

Published in 2001, this is the oldest of the three documents mentioned in this paper, but has many of the same strengths. Indeed, it was one of the first knowledge management documents to capture the practical experience of a wide variety of practitioners, and highlight emerging trends.

It has good material on:

- Knowledge management case studies
- Alternative knowledge management practices and resources
- Managing the risks involved in knowledge management, and implementing policies, standards and guidelines to avoid those risks.

All three guidelines provide a welcome bridge between the high-level ISO 9001:2000 framework and practical activities in the work place.

4. An Emerging Knowledge Management Strategy

Is there a widely accepted knowledge management strategy?

Since knowledge management began to emerge as a discipline in the 1990s, two main strategies have been proposed: the first focused on collection, storage, and reuse of explicit knowledge in documents and IT systems. It was a kind of "knowledge warehouse" approach.

The second strategy, which emerged in reaction to the perceived failures of warehousing, focused on connecting people to people, where knowledge management was seen mainly as a social communication process with emphasis on tacit (personal) knowledge.

Today, knowledge management is generally seen as a customized blending of these alternatives within specific organizational contexts, and concentrates on achieving an appropriate balance among four elements: people, process, technology, and content.

As mentioned earlier, ISO 9001:2000 encourages continuous person-to-person contacts as a way of overcoming communication barriers and facilitating knowledge flow. This is framed as a need for contact with customers, both internal and external. It is also about enhancing internal networks by developing, for instance, collaboration tools and communities of practice.

Information technology (IT) is certainly important to knowledge management. It may be used, for instance, to capture process-based activities and record them in a Quality Manual. However, it should not used to warehouse every scrap of information available within the organization.

In the emerging KM environment, information technology serves as a means to:

- Provide people performing tasks with convenient access to reference materials and job aids geared to their needs;
- Support on-line networking; and
- Deliver self-paced, just-in-time training.

6. Conclusion

What key messages can be taken from this discussion?

First, the ISO 9001:2000 environment is very conducive to knowledge management.

Prime opportunities for knowledge creation and transfer in this environment include:

- Interactions with “customers”;
- Process mapping and continuous improvement activities;
- Collegial audits;
- Direct colleague-to-colleague knowledge transfers by means of coaching, mentoring, job shadowing, communities of practice, informal debriefings, etc.;
- A thoughtfully developed Quality Manual;
- An on-line repository for useful reference materials; and
- A balanced training program incorporating both face-to-face and technology-assisted methods.

There are real benefits to be gained by making knowledge management part of the fabric of everyday business. By no means the least of them is a greatly reduced need for inefficient knowledge end-of-career knowledge capture exercises. The ISO 9001:2000 Standard provides an excellent platform from which to accomplish this important function.

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Safeguards Culture: Lessons Learned

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Abstract:

At the 2005 INMM/ESARDA Workshop in Santa Fe, New Mexico, I presented a paper entitled "Changing the Safeguards Culture: Broader Perspectives and Challenges." That paper described a set of theoretical models that can be used as a basis for evaluating changes to safeguards culture. This paper builds on that theoretical discussion to address practical methods for influencing culture. It takes lessons from methods used to influence change in safety culture and security culture, and examines the applicability of these lessons to changing safeguards culture.

Keywords: safeguards; culture; influencing; safety; security

1. Introduction

At the 2005 INMM/ESARDA Workshop on "Changing the Safeguards Culture: Broader Perspectives and Challenges," in Santa Fe, New Mexico, I presented a paper entitled "Changing the Safeguards Culture: Broader Perspectives and Challenges." That paper, co-authored by Karyn R. Durbin and Andrew Van Duzer, described a set of theoretical models that can be used as a basis for evaluating changes to safeguards culture. This paper updates that theoretical discussion, and seeks to address practical methods for influencing culture.

2. Background

This paper takes lessons from methods used to influence change in safety culture and security culture, and examines the applicability of these lessons to changing safeguards culture. Implicit in this discussion is an understanding that improving a culture is not an end in itself, but is one method of improving the underlying discipline, that is safety, security, or safeguards. Culture can be defined as a way of life, or general customs and beliefs of a particular group of people at a particular time. There are internationally accepted definitions of safety culture and nuclear security culture.[1] As yet, there is no official agreed upon definition of safeguards culture. At the end of the paper I will propose my definition.

At the Santa Fe Workshop the summary by the Co-Chairs of Working Group 1, "The Further Evolution of Safeguards," noted: "It is clear that 'safeguards culture' needs to be addressed if the efficiency and effectiveness are to continue to be improved. This will require commitment and change at all levels, from States to facility operators. Cultural change has to come from good leadership, doing the right thing and 'beliefs' are not sufficient – behaviour is what counts. We are optimistic that with sufficient effort and the right incentives, change can be accomplished quickly."

[1] Safety Culture: That assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance. (IAEA INSAG-4)
Security Culture: The assembly of characteristics, attitudes and behavior of individuals, organizations and institutions which serves as a means to support and enhance nuclear security. (IAEA Nuclear Security Series No. 7)

3. Models

In Santa Fe, we stressed the theoretical basis for culture improvement through the use of social science models. These models help frame the discussion of organizational culture. Most notable is Schien's model[2] in which the beliefs of an organization serve as a foundation, what an organization says about itself is in the middle, and what it actually does, and can be measured is at the top. A second model considers a hierarchical structure. In this approach the actions of the individuals in an organization are assumed to be influenced by the policies established at the top political level, and effected through the actions of management and organizations.

In 2005 much of the discussion of safeguards culture focused on the differences between the traditional verification activities of the IAEA and the new requirements of the Additional Protocol and Complementary Access. Without judging how successful the changes related to this aspect of safeguards have been, I would note that the emphasis in the safeguards community seems to have shifted from a focus on this aspect to a discussion about 3S, that is, the integration of regulatory approaches to safety, security, and safeguards in the nuclear field. While many are sceptical of this approach, there may be value in considering the cultures of the three components to see what lessons can be learned from some to apply to others. In this case, the experiences in achieving enhancements in safety culture and security culture may have some relevance to the enhancement of safeguards culture.

The idea of 3S is that there is organizational overlap and synergy among safety, security, and safeguards. This is usually expressed by a Venn diagram that shows exclusive areas of activity and areas of intersection. For example, fire safety might be exclusively in the safety circle, while physical protection might be shared by security and safeguards. And access control would be shared by all three. So in this concept where would we place safety culture, security culture, and safeguards culture?

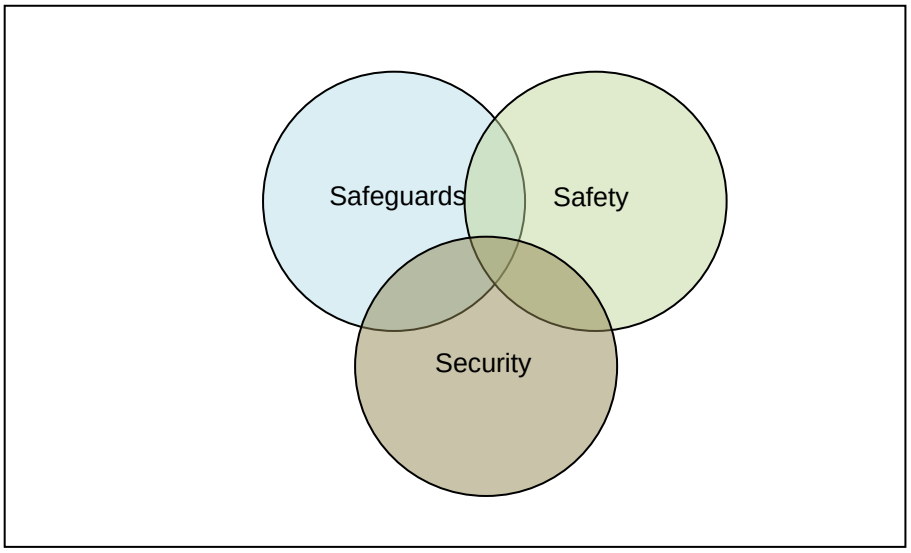


Figure 1: 3S

In fact, all three cultures exist simultaneously, separately, and yet they have similar characteristics. A strong safety culture will help prevent accidents. A strong security culture will help prevent theft or diversion of nuclear material by non-state actors. A strong safeguards culture will help prevent unauthorized use of nuclear material by state actors. We could superimpose an elevated set of circles that might also intersect, but with each culture leading to improvements in the underlying discipline.

Or maybe the 3S model is pertinent, but lacks a fourth circle. Let's call it Mission. After Chernobyl, some nuclear power station operators resisted a call for increased emphasis on safety because it was thought to detract from the mission. In fact, as the emphasis on safety became instilled in plant

[2] Shein's Model is appended at the end of the paper.

operators, and safety metrics improved, the metrics associated with mission (such as fewer unplanned outages and increased capacity factor) improved as well. It seems that, over time, the cultural norms become embedded into the mission. In our experience with a pilot project on nuclear security culture in the Russian Federation, so far we have only anecdotal evidence, but the experience appears to be similar. For example, some facilities that have incorporated a new emphasis on security culture report a reduction in the number of security incidents. Together with our Russian colleagues we have developed a set of nuclear security culture metrics that are being tested and refined.

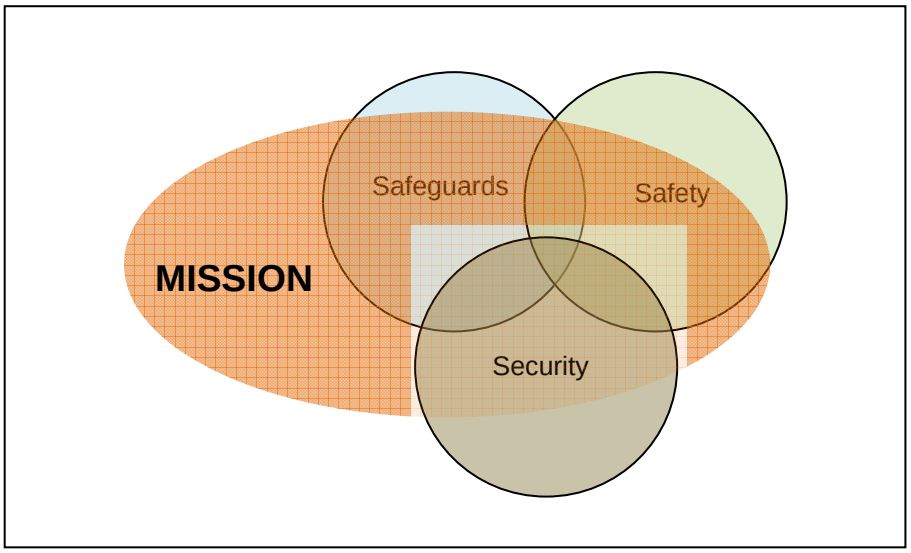


Figure 2: 3S plus Mission

4. Development

The IAEA has developed guidance documents for safety culture and nuclear security culture. So far, a safeguards culture guidance document has not emerged. If such a document were to be written, it should draw upon the experience of the others. In our Santa Fe paper we suggested that the first step in promoting safeguards culture is propagating a strong understanding and awareness of what safeguards culture is, and why it is important. The second step is gaining a strong commitment from those involved in the safeguards process from manager to implementer. In considering how to develop a program to improve safeguards culture, the experience of developing programs to enhance nuclear security culture suggests an approach consisting of a series of steps:

1. Identify the target audience for the program.
    a. Within the host country:
        i. Government
        ii. Other political entities such as regulatory bodies
        iii. Nuclear industry including the nuclear facilities design community
        iv. The military
        v. Individuals working at nuclear facilities
    b. Within international organizations
        i. United Nations Security Council
        ii. IAEA Board of Governors
        iii. Representatives of national governments
        iv. International civil servants
        v. IAEA inspectors
2. Conduct a baseline analysis to determine the directions of the program
    a. Internal to states – Is there a strong understanding within the state as to what safeguards culture is, and why it is important?
    b. Internal to the IAEA – Is there a strong understanding within the IAEA of safeguards culture, and why it is important?
3. Attain top level management support for implementing change

4. Provide for a regulatory basis for the program
5. Incorporate lessons learned into a formal feedback mechanism
6. Develop educational and training programs to highlight the importance of the program

During a recent Safeguards by Design workshop sponsored by the IAEA one of the recommendations was that a safeguards culture should be fostered in the facility design community. The report of the conference noted that unlike safety culture which is strongly supported, safeguards is little-known to many designers. It was suggested that increased awareness and a better appreciation of safeguards goals and requirements be created through a proactive outreach initiative sponsored by the IAEA and endorsed by member states. It cited ESARDA, among others, as a possible venue to disseminate safeguards culture information to designers.

As we pointed out in our Santa Fe paper, influencing a culture by establishing new norms requires long-term planning and patience. Here we have tried to identify some principles that have proven themselves in the enhancement of other kinds of culture, and suggested that these principles could be applied in changing safeguards culture.

5. Challenges

There are two main challenges to developing a program to improve safeguards culture. First is that there are so many actors involved in the safeguards arena. Second, the hierarchy of responsibility is not so clear-cut as in the safety and security realms. Therefore, the overarching challenge is to develop a program to improve safeguards culture, identifying who should apply the principles, and who leads the change. Because safeguards culture might be in conflict with a national security goal in some countries, it is necessary to have buy-in at the highest national levels for a safeguards culture to take hold.

In identifying actors, individual national regulatory bodies would seem to be important actors. Regional organizations like Euratom or ABACC also have a role. At the IAEA, leadership needs to come from the Board of Governors. But the impetus for that leadership depends on national representatives.

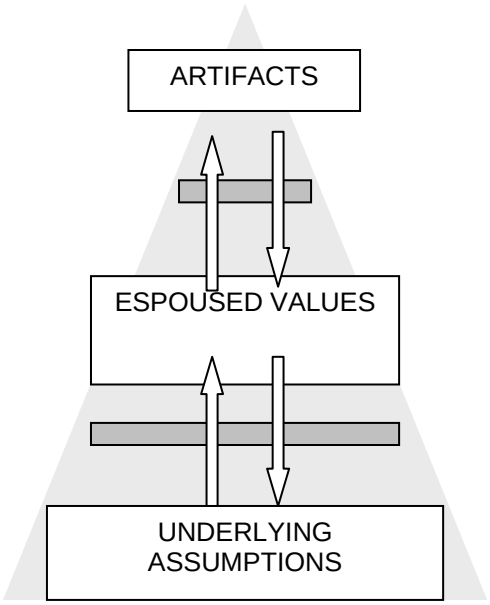
At the facility level, a positive safeguards culture implies active support for the concept of international safeguards. If operators accept the importance of their efforts in preventing nuclear proliferation, they are likely to be more accepting of the burden that safeguards places on their operations. National and regional regulatory authorities can contribute to that aspect of safeguards culture by making it an important part of their regulatory oversight.

6. Definition

Based on lessons learned from safety and security culture I will end the paper by proposing a definition for safeguards culture. Safeguards culture has been a frequent topic of discussion over the years, and can benefit from an agreed upon definition. Here is my version, designed to further the discussion, and lead to the adoption of a version that we all can support, to help enhance the effectiveness of international safeguards.

6.1. Safeguards Culture

A shared belief among individuals, organizations, and institutions that international safeguards is an important undertaking to prevent the proliferation of weapons usable nuclear material, which is manifested by strict attention to safeguards requirements and affirmative cooperation with safeguards authorities.



THE THREE LAYERS OF CULTURE

Figure 3: Shien's Culture Model[3]

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3S-CBRNE and information exchange: considerations on the need to know

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Abstract:

The 3S-counter-CBRNE effort involves several authorities in a country, each linked to their respective international networks for information exchange, relevant to their area of expertise. All of these authorities must be able to trust that the others are in a position to share with them the information the authorities need in order to do their work, and they must know which information is of relevance to each of them—what each of them needs to know.

The need to improve information exchange is a recurring theme in international meetings and has resulted in new initiatives for new international information exchange systems. The multiplication of information exchange systems, however, will not solve the perceived problems. On the contrary, undue multiplication of reporting channels and separation between information systems leads to unnecessarily complex reporting schemes and additional work in trying to ensure that appropriate parties receive the appropriate information.

Problems caused by incomplete reporting to, lack of use, or suboptimal features of existing information exchange systems are dealt with most efficiently by improving the reporting, the use and the features. Moreover, we must consider the system as a whole so that the improvements may profit both the national and the international dimension of information exchange.

The paper discusses features of the present information exchange and possible reasons for the perceived imperfections, from the 3S regulatory authority's point of view.

Keywords: 3S, CBRNE, information, exchange

1. Introduction

Justifiable use of nuclear energy and radiation requires that we undertake to protect people, society, the environment, and future generations from the harmful effects of radiation. This mission is accomplished by implementing a system of control measures for safety, security, and safeguards—the 3S. The responsibility of implementing such a system is shared by the international community, national legislators, competent authorities, and the users of nuclear and radiological technology. The 3S system interfaces with the counter-CBRNE effort: deterrence, prevention, and detection of and response to illegal activities involving chemical, biological, radiological, or nuclear materials (and explosives), led by law enforcement.

Credibility of the 3S-counter-CBRNE effort requires trust. Trust between the different authorities, trust by the people in the state, trust within the international community. Trust can be built through effectiveness, efficiency and a certain degree of visibility in the measures taken to ensure nuclear and radiological safety, security, and safeguards. A key characteristic of a credible national system is a coordinated trusted community of authorities (intelligence, defence, law enforcement, regulatory authorities), who each belong to international networks in their own area of expertise, connecting the national level into the international community. To achieve

effectiveness and efficiency, the authorities must communicate well: our information exchange policies and practices both on the national and on the international level, as well as between the two levels, should provide the right information at the right time to the right user—the one with the need to know. And the policies and practices should be agreed between and known to the members of the 3S-counter-CBRNE community.

2. The need to know: national level linked to international networks

The three S's are largely synergistic—a great proportion of the control measures for each S contributes directly to one or both of the other S's—but some requirements of the S's conflict with each other. Safeguards and security share a common motivation and most of their methodology for securing the nuclear material; safeguards are designed to act as a deterrent through prevention and detection of potential diversion of materials from legal, licensed use. Thus, for the purposes of this paper, safeguards and security are one and the same. Between all three S's, we must coordinate ourselves to take advantage of the synergies and to avoid the problems of conflicting requirements, hence the 3S approach. Although this paper deals mainly with security, we wish to maintain that idea of coordination.

One example of the conflicting requirements between safety and security is information on the nuclear and other radioactive materials and on the lines of defence in protecting the people and the material. In safety, that information is shared extensively. In security, information is shared on a need-to-know basis. The difference in approaches is due to a difference in the threat: the control measures of safety are designed against a passive chain of events and the control measures of security are designed against an active force. In an accident, complete transparency should help limit the consequences and help proceed to safe state in a predictable way. For security it could have the opposite effect, as the adversary would be expected to take advantage of the available information. The fact that security information is, inherently, restricted on a need-to-know basis presents a challenge to the requirement of trust. We must be careful with classification of information: define what must be protected, protect it, and make all else public.

The 3S-counter-CBRNE work involves several authorities in a country, each linked to their respective international networks for information exchange, relevant to their area of expertise (Figure 1). All of these authorities must be able to trust that the others are in a position to share with them the information the authorities need in order to do their work, and they must know which information is of relevance to each of them—what each of them needs to know.

Measures taken to ensure security of nuclear and other radioactive material and the related facilities are based on the properties of the materials and facilities themselves and to the threat against which they are to be protected, as recommended in international references [1, 2]. Design basis threat (DBT) is a commonly preferred approach [3]. The state DBT defines the threat that the security measures shall be effective against, and is developed in cooperation by competent authorities, including the intelligence community and the 3S regulatory authority. The regulatory authority sets the requirements for nuclear security, oversees the implementation, and evaluates the effectiveness of the implemented measures. In order to do this it needs information on the motivation, intentions, and capabilities of the actors presenting the threat, in the DBT development and review process and whenever there are shorter-term changes in the threat. This information falls within the competence of the intelligence and law enforcement organisations to produce and to provide. There should be an established mechanism to maintain the DBT and to communicate the shorter-term changes in the threat to the relevant parties—the ones with the need to know.

Threat assessment is an important interface between the 3S authority and the rest of the counter-CBRNE community. Another one is operational cooperation: expert advice in radiation detection technology and radiation safety during field operation and assessment of measurement results.

Peer-to-peer exchanges, for example between the regulatory authorities, between the nuclear operators, and between these two groups, are vital for continuous development and improvement of the 3S. Sharing of operational experiences in security and sharing of information related to performance of the technical elements of nuclear security systems presents a particular challenge. How to ensure both information security and an enriching information exchange environment? The mechanisms are not as straightforward for

security as they are for safety, and they remain a challenge.

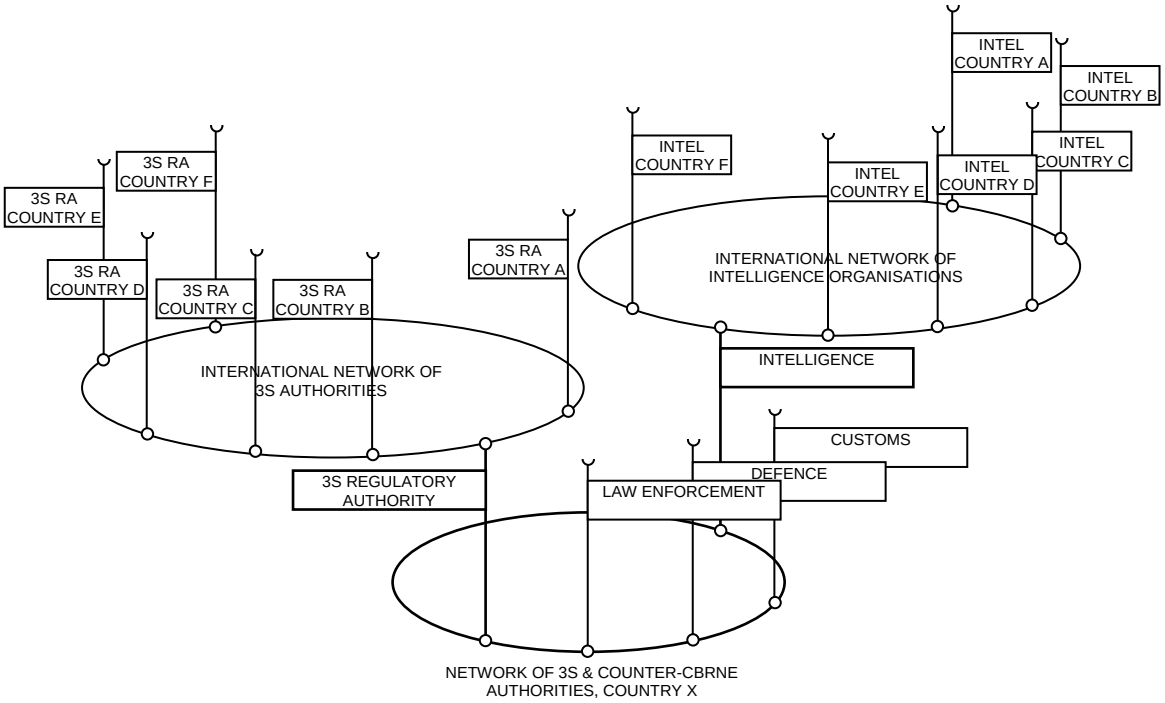

Figure 1: National 3S and counter-CBRNE authorities' communities are linked to international information exchange networks through their member organisations. The linkages and the rules of information flow should be known to the parties involved in the networks.

3. Existing systems: effective and efficient?

The 3S regulatory authorities are members of several international information exchange systems, most of which are hosted by the IAEA. States report to the international community on events that have or may have radiation safety consequences, i.e. on nuclear and radiological emergencies and incidents. The reporting policies and thresholds are defined in international conventions [4, 5]. The combined Early Notification and Assistance Conventions notification system is hosted by the IAEA Incident and Emergency Centre (IEC/ENAC) [6]. Within the European Union there is a similar system, the European Community Urgent Radiological Information Exchange (ECURIE) [7]. In addition to the emergency systems, the IAEA hosts information sharing and analysis tools, for example, the IAEA Illicit Trafficking Database (ITDB) [8] and the Nuclear Events Web-based System (NEWS) [9], linked to the International Nuclear Event Scale (INES). Through the former the member states share information on illicit trafficking, thefts and losses, and other acts involving radioactive materials not in accordance with the national regulations. Malicious Acts Database (MAD) [10] is a recent addition. Within the nuclear safeguards regime, states report on nuclear fuel cycle related research, development, and trade [11]. In all likelihood this list is not exhaustive, and the IAEA and other international organisations also gather information from open sources. Information tools are likely to become more and more important in the effort to manage the expanding nuclear scene, not to mention the counter-CBRNE effort. The data in itself gathered in the various information systems is of limited value; in the interest of added value emphasis should be placed on analysis of the data and on reporting the analysis results.

The present policy and practice in Finland of notifications on nuclear and radiological emergencies and incidents, implemented by STUK (Radiation and Nuclear Safety Authority), is based on international conventions (ENAC), European Union legislation, and bilateral agreements developed over the years with several countries. In the scheme describing the policy and practice

there are 72 event categories, grouped by type of facility (NPP, research reactor, other), type of activity (use of nuclear energy, use of radiation), type of material (nuclear and other radioactive, category of material), cause of event (accident, malevolent activity), and the consequences or potential consequences, locally, for Finland, and for other countries. There are three groups of information exchange partners: (1) national authorities, (2) authorities in the Nordic Countries, and (3) IAEA (IEC/ENAC), EC (Ecurie), and authorities in Germany, Russia, Estonia and Ukraine. The scheme specifies which group is informed of which events, and how. Bilateral agreements are made to complement international conventions. They may have considerably lower notification thresholds and they may define additional responsibilities for the parties to the agreements. In addition to the emergency notifications, Finland is a member, through STUK, of the NEWS/INES and ITDB information exchange systems and implements nuclear safeguards reporting in accordance with the Additional Protocol [11] and Integrated Safeguards.

Some of the cases in the IAEA information exchange systems are of interest not only to the 3S regulatory authorities but also to other counter-CBRNE authorities, and the 3S authority should have a mechanism to relay the relevant information to them. The intelligence and law enforcement authorities have their own international information exchange networks and protocols. They should be able to tell which information is of relevance to the 3S authority and to distribute it accordingly. This may include, for example, selected reports of the Europol. Interpol hosts an illicit trafficking database, project Geiger [12], which overlaps with and complements the IAEA ITDB. The cooperation between the two organisations aims at improved analysis of trends, risks, threats, routes, and methods. It is important that the results of the analyses be shared within the 3S-counter-CBRNE community in a systematic manner known to the relevant recipients of the results.

4. Goal: coordinated approach?

The need to improve information exchange is a recurring theme in international meetings and has resulted in new initiatives for new international information exchange systems. The multiplication of information exchange systems, however, will not solve the perceived problems. On the contrary, undue multiplication of reporting channels and separation between information systems leads to unnecessarily complex reporting schemes and additional work in trying to ensure that appropriate parties receive the appropriate information.

Why should one feel there is not enough of information sharing going on? A few possible explanations for this perception come to mind:

- Lack of information is misinterpreted as lack of communication, because rules of information flow are not known to interested parties. As an example, if the 3S regulatory authority is not receiving regular updates on the state threat assessment, it may interpret it as reluctance to share information, when in fact the reason for the silence may be that there is no new information to share. The problem is solved by following an agreed information exchange and update procedure and ensuring that the members of the national 3S-counter-CBRNE community know what information is relevant to each organisation.

- Imperfect reporting to international information exchange systems impairs the quality of the analysis of threat, trends, and risks. Incomplete or varying reporting by member states may be partly due to misunderstanding about the scope of a system and partly due to the perception that the system does not provide added value. For the sake of added value it should be ensured that the information flow is two-way—that information is not over-classified, that the scope is clear, and that the national points of contact are in a position to access and distribute efficiently the analysis results (the actual product of the system) in the member state.

- Uncertainty and vagueness of classification of information may prevent distribution unnecessarily. It has been known to happen that an international source of information has intended the information to be distributed through the national contact points to “competent authorities”, but as the interpretation of the term competent authorities varies from one area of expertise to another, the original distribution has not reached all intended recipients.

- Sometimes it is most difficult to see close up. As the international meetings under the heading of CBRNE typically are quite multidisciplinary, it is sometimes possible to learn new things about the policies and

practices of one's own country by participating in an international meeting.

- Open-source information in high-quality media may at times be better than the information through official channels. Efficient media is a good thing, for the public has a need to be informed. What is not so good is to notice that some incidents apparently are reported in open-source media only—or that they are reported very much later in the official channels.

With the creation of the IAEA IEC, the IAEA NEWS system has become part of it, and the IAEA is developing a single system to incorporate both of these [6]. The new MAD database has also been presented as part of a development towards an integrated platform [10]. Further, we understand that the IAEA and the European Commission are endeavouring to link the IEC/ENAC and the Ecurie together in such a way that the member states need to report to only one of them. These developments are most welcome.

Participation in the information exchange systems is based partly on conventions and law and partly on voluntary contributions. Irrespective of the motivation, we should ensure that the systems create added value for the participants. This is expected to require considerable coordination and sharing of information between the systems, if not a total fusion. Interactions and comparisons between the various international information systems should result in aggregate analysis results distributed systematically to the participants. The information systems should help sharing of operational experiences and best practices. They should increase our understanding of the global state of the 3S and the counter-CBRNE work and, ultimately, help us improve it.

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Challenges for education & training for nuclear safeguards and non-proliferation

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Abstract

The original pessimism of one decade ago with great concern of vanishing nuclear knowledge is turning around with the several international education and training initiatives in the nuclear field. Moreover past lessons learnt resulted in the concept of including also non-proliferation by reactor design. Nuclear safeguards is a supranational issue, for which training remained a niche for the Joint Research Centre (JRC) in Europe and even with limited competition worldwide. The JRC has more than 30 year experience in providing Nuclear Safeguards courses to inspectors. Moreover the lack of safeguards in the academic curriculum of nuclear students is recently tackled by the European Safeguards Research and Development Association (ESARDA) and filled with a course hosted at JRC. For the last five years, as part of the international effort to promote nuclear knowledge, ESARDA's Working Group Training and Knowledge Management has been educating more than 250 young professionals and students about nuclear safeguards and non-proliferation and aim to complement nuclear engineering studies by including nuclear safeguards in the academic curriculum. The weeklong courses, with lectures from some of the leading experts in the field of nuclear safeguards, are open to masters degree students, in particular nuclear engineering students, but also to young professionals and those studying international relations and nuclear law. Fostering a two-fold technical-juridical education in the EU is compliant with one of the original roles of JRC described in the EURATOM Treaty and fits with the current view of the International Atomic Energy Agency. In order to reinforce the dwindling teaching capacity, in particular also in nuclear safeguards and non-proliferation, international collaboration is fostered with a new Working Group NUSASET (Nuclear Safeguards and Security Education and Training).

1. Introduction: a historical record

Nuclear knowledge originates mainly from European scientists in the beginning of the 20th century, with amplified investigations in the thirties when the potential of nuclear energy applications was discovered. With the timely coincidence of World War II, military research programmes focused on non-peaceful applications of nuclear energy with success. Only after Eisenhower's "Atoms for Peace" speech in 1953, the technology needed for civil use of nuclear energy was transferred and research focused on commercial reactors.

Commercial reactors were established fast on national level, whereas for the supporting fuel cycle activities a supranational approach remained welcome. The nuclear material was controlled in the European territory by the EURATOM Treaty, but in 1968 new international regulations following the Non Proliferation Treaty were faced. The EURATOM inspectorate wanted to be trained by JRC for controlling the nuclear material flow and nuclear activities while receiving international recognition for that. Contrary to the safeguards, safety remained also after the increased attention in 1979 with the Three Mile Island accident addressed with national training of operators and inspectors. Two decades passed with a stable nuclear energy but a decreasing public acceptance (in particular after the Chernobyl accident of 1986). The interest for nuclear studies dramatically decreased in the nineties, and nuclear education was abandoned by many engineering faculties.

A declining knowledge and expertise in the nuclear field were reported e.g. by the OECD in 2000. [1] The International Nuclear Safety Advisory Group to the IAEA Director General emphasized in Note No.4 of 2001 the importance of maintaining capabilities for nuclear research and education [2]. In 2002, a report of the Secretary-General of the United Nations [3] underlined that "there has never been a greater need for education in the areas of disarmament and non-proliferation" and that "Additional disarmament and non-proliferation related educational efforts are needed at all academic levels, for which support by the UN and its Member States is crucial". It is also clear from the time-period of the major civil nuclear developments in Europe that a generation change has occurred very recently, causing a lot of know-how to disappear.

For nuclear technology, which involves many disciplines of science and engineering, knowledge is one of the most important resources and needs to be managed carefully. Three different groups need to be addressed differently: (1.) students finishing off their specialization in the nuclear field: this group forms the pool from which the incoming generation can be recruited; (2.) experts: the growing generation, enlarging their experience, broadening their field; (3.) the retiring generation, of which as much expertise should be recuperated.

The first group of students needs education; the second group training; and for a well-planned change in generation with the third group knowledge management is crucial.

2. The European dimension

Already the origin of the JRC as so-called "Joint Nuclear Research Centre" in Article 8 of the Euratom Treaty includes not only "research programmes" in civil (and safe) use of nuclear fission energy, but also the establishment of "a uniform nuclear terminology and standard system of measurements"[1]. In line with this, JRC's second programme (1963-1967) focused on documentation, information and training. [4]

Safeguards training was developed mainly in the early seventies, after a new set of regulations on nuclear material control were drafted in line with the Non Proliferation Treaty of 1968. A first two-weeks course on Safeguards and Fissile Material Control took place at Ispra in March 1975. The course addressed (i) safeguards regulations with lecturers from the Directorate of Euratom Safeguards, (ii) the Non-Destructive and Destructive Analysis and Containment/ Surveillance techniques with lecturers from JRC and (iii) plant-specific implementations mainly in the front-end of the fuel cycle, lectured by industry experts (BNFL, Alkem GmbH, NUKEM – Hanau, and RCN – Petten). In the late seventies the JRC Ispra was recognized as a real training centre for inspectors and operators. [12] After having trained many key-personnel in nuclear industry and inspectorates, the safeguards training decreased with the diminishing demand.

In the nineties the JRC retook training with the increasing importance of the back-end of the fuel cycle, in particular to address safeguards implementation in the reprocessing plants of La Hague and Sellafield. Whereas the Ispra site used further its large expertise in Non Destructive Analysis (NDA) techniques and process monitoring, the Karlsruhe site was, in particular with the On-Site Laboratory (OSL) project for La Hague and Sellafield focusing on Destructive Analysis (DA) techniques.

Article 9 of the Euratom Treaty declares that within the framework of the Joint Nuclear Research Centre "schools for the training of specialists, particularly in the fields of prospecting for minerals, the production of high-purity nuclear materials, the processing of irradiated fuels, nuclear engineering, health and safety and the production and use of radioisotopes" are to be set-up and "an institution of university status" is to be established. this was enabled only about 40 years later. A transparent exchange mechanism between the different national higher education systems had to be agreed between leading universities in Europe. This was obtained within the Sorbonne-Bologna process in 1998, which formalized a European Credit Transfer System (ECTS)[2] for studying at those European universities. With the decreasing number of nuclear engineering and nuclear physics students, the number of nuclear courses offered today are more

[1] Although a common standard was originally felt most needed for radioprotection and environmental radioactivity, it was more generally meant as a central certification lab for nuclear measurements.

limited than 30 years ago, but they are with the ECTS internationally recognized and therefore more accessible for foreign students.

The European Commission replied to the dwindling teaching capacity in nuclear science and technology by financing the set-up of European nuclear higher education in a sustainable manner. The temporary European Nuclear Engineering Network, established through the EC 5th Framework programme project, was given a permanent character by the foundation of the European Nuclear Education Network (ENEN) Association in 2003, pursuing a pedagogic and scientific aim. [5] In follow-up of the Sixth Framework Programme of the European Atomic Energy Community (EURATOM) the Council Decision of 18 December 2006 concerning the Seventh Framework Programme (FP-7) of EURATOM for nuclear research and training activities identifies education and training concerning nuclear safety and radiation protection among its priority areas. Moreover the EURATOM FP-7 in nuclear fission pursues a holistic strategy, entangling research & development & deployment with education & training. [6]

Nowadays the ENEN includes thirty-seven effective university members providing ENEN education under the coordination of the CEA-INSTN research institute and with the support of five other research centres. The network has been growing with the support of nuclear industry, regulators and research centres and counts in addition fifteen associated members. In total eighteen different EU Member States are represented. The activities of the ENEN Association are organized in five committees: the Teaching and Academic Affairs Committee, the Advanced Courses and Research Committee, the Training and Industrial Projects Committee, the Quality Assurance Committee, and the Knowledge Management Committee. The ENEN aims to provide a common qualification in nuclear engineering, with a mutual recognition and with a facilitated mobility of teachers and students. [7]

Over these five years, the ENEN has completed a variety of tasks and delivered appreciated products to the European Higher Education by harmonizing nuclear education. Every year successful students are granted a Master of Science degree in Nuclear Engineering (EMSNE)[2]. The students with an ENEN diploma are highly valorized in the reactor industry. The Belgian Nuclear higher Education Network (BNEN) pioneering project for inter-university Nuclear Engineering Education was successful partially thanks the Belgian nuclear industry, requiring all junior management staff to achieve a BNEN degree.

Nowadays the ENEN Association intends to expand its activities with professional training programmes, directed to key functions in nuclear industries, regulatory bodies and nuclear applications to harmonize the existing training programmes and to promote their international mutual recognition.

JRC's 7th FP (2007-2013) is drawing again more attention to education and training in the nuclear field, and explains close collaboration with the ENEN, laid down in a memorandum of understanding.

3. Current Education on Nuclear Safeguards and Non-Proliferation in the EU

Education is commonly defined as a basic, knowledge-driven, learning process, involving academic institutions as suppliers and students as customers, which encompasses the need to maintain completeness and continuity of competences across generations.

3.1. ESARDA's Education in Nuclear Safeguards and Non-Proliferation

With the support of ESARDA, in particular the ESARDA Working Group on Training and Knowledge Management, the JRC is organizing yearly a Nuclear Safeguards and Non Proliferation course. This course was this year organised for the fifth time and attended by 55 students, mounting up the total number of students that followed this course to almost 300 [8, 9]. The BNEN/ENEN has recognized the course as a standard academic one semester course of 3

[2]EMSNE with ENEN certification is documented on the ENEN website

ECTS[3]. To formalize and maintain this recognition a course syllabus is developed with a chapter dedicated to each standard issue of the course: i.e. historical background, nuclear materials, nuclear material management principles, overview on treaties, the NPT and safeguards agreements, safeguards under the EURATOM Treaty, statistical aspects, random sampling, destructive sample analysis, non-destructive assay, containment and surveillance, nuclear forensics, and remote environmental sampling. The topical lectures are excluded, but the participants receive a handout of the presentations.

With the contacts made at this course, some students profit from a JRC traineeship to gain some working experience of working in a multinational research environment, which can even be qualified for credits as part of a degree course. It benefits the JRC by providing extra resources, an influx of fresh ideas and closer links to the academic community.

3.2 . Reinforcing Teaching Capacities by International Collaboration

At the international Safeguards Workshop, co-organised by the INMM and ESARDA in Tokyo, October 2008 an international Working Group for Non-Proliferation and Safeguards Education & Training was established. Even though informal, the group (with members from PNNL, LANL, IAEA, MEPHI, Texas University, Monterey, JRC, etc.) committed to tackle a.o. the following issues [10]:

1. RECOGNITION of required safeguards competence (also in career paths): In all nuclear countries, nuclear engineers can graduate without being educated on safeguards and non-proliferation. In virtually all countries this is the rule rather than the exception. Specialist training programs for safeguards and non-proliferation are serving current needs (US, EU, RF), but more needs of government, industry and NGO's can be identified. Career paths in the safeguards/non-proliferation field are not so easily associated with regulatory requirements for education&training as is the case in nuclear safety / security. This complicates the tailoring of education&training programs to specific needs. Institutions like INMM and ESARDA need to develop institutional ties to other organisations such as ANS and ENS to help them to appreciate the importance of safeguards and non-proliferation in achieving their mission.

2. More active RECRUITMENT and better utilisation of education programs (by expanding capacity). There are robust and successful programs to serve as models in education in both policy and technical areas (MEPHI, TEXAS A&M, ESARDA, Monterey, Georgia etc) but they are rarely addressed together in a balanced way. Outreach programs are needed to introduce and give a general base of knowledge to a broad cross section of university students and young professionals to make them aware of safeguards/non-proliferation as a career option. The same can be said about current nuclear workforce and the need to increase awareness of safeguards/non-proliferation. Outreach is needed to potential employers to bring them on board as stakeholders in safeguards/non-proliferation matters and the associated education&training. (An example of successful outreach programs for both technical and policy students is the US Next Generation Safeguards Initiative). Education&training programs must stay abreast and on the cutting edge of latest technological developments and applications (also outside the nuclear area). This is another reason why links with research institutions is deemed highly valuable. Dissemination and awareness raising to students and young professionals could be done via an information portal. Such website, for use by students, teachers and interested professionals, could bring significant added value in areas of job descriptions, career profiles, job opportunities, studying opportunities, reference materials, workshops/conferences.

[3]ECTS: European Credit Transfer System as defined under the EU's Erasmus/Socrates programme for education and training

In Ispra, April 2009, a follow-up of this workshop yielded a more formal creation of this working group which received the acronym NUSASET :"Nuclear Safeguards and Security Education and Training". The scope of this working group was summarised in Annex 1 and working procedures were drafted, in preparation of the stakeholders meeting in Tucson (July, 2009). [11]

4. Training in nuclear safeguards and security offered by the JRC

Training is commonly defined as an application-driven learning process, involving industrial training organisations as suppliers and professionals as customers. This schooling activity goes beyond the regular academic education scheme of more generic nature as it learns a particular skill required to deliver a particular outcome.

4.1. In-house training for Europe

JRC has many years of tradition in providing training to inspectors, originally in support to EURATOM – DG TREN and nowadays extended to the IAEA, and even law enforcement bodies. For safeguards, an international issue, JRC's neutral and central role for providing such training was accepted, while for safety of nuclear installations, an issue under the responsibility of the Member States, national training was put in place. The training courses given at JRC are listed with a small description in Annex 2.

Laboratories, at the Nuclear Security Unit in Ispra, mainly PERLA (Performance Laboratory) but also TAME (Tank Measurement laboratory), have been put at the disposition of inspectors for testing/calibrating instruments and for training. There are nowadays at Ispra yearly about sixteen weeks of training courses offered to DG TREN and IAEA inspectors. This set of courses was since 2008 extended upon request of the IAEA with courses on Complementary Access and Design Information Verificationand Advanced NDA techniques.

At the DA Laboratories in Karlsruhe specialized courses on the use of some instrumentation (incl. software) as in place at the OSL, e.g. on the Hybrid K-edge (for measuring the concentration of U and Pu of an input solution at a reprocessing plant) are organised upon request. In the mid nineties, Karlsruhe's Nuclear Chemistry Unit took the opportunity to profile itself also as a nuclear forensics laboratory. With the experience gained in the analysis of seized nuclear material (such as the Pu in 1994 at the Munich airport), training on the response to illicit trafficking cases and analysis of seized material was setup. Nowadays ITU provides yearly four courses on nuclear forensics to regulatory authorities (Customs officers), law enforcement bodies (Police authorities) and measurement laboratories, partially in collaboration with the IAEA and with the Advanced Training Centre of the national Karlsruhe Research Centre. Two types of training are given: (i) the Customs and Police authorities are trained on the Model Action Plan i.e. on the different actions to be undertaken at the detection of an illicit trafficking case (specified under the Model Action Plan), whereas (ii) national analysts of measurement laboratories are taught the different aspects of the nuclear forensics science.

4.2. *Transmitting safeguards training to the Commonwealth of Independent States*

With the addition of nuclear safeguards to the TACIS[4] programme in 1994, the EC approach was based on a sustainable improvement of nuclear security in CIS with as a first pillar the provision of training on safeguards methodology. End nineties, several projects in the Moscow and Chelyabinsk regions lead to the setup of a large training centre, the Russian Methodological Training Centre (RMTC) at the research centre IPPE in Obninsk and a smaller one, the Ural Siberian Methodological Training Centre (UrSiMTC) at the research centre in Snezinsk. Whereas RMTC grew to a well-established training centre, UrSiMTC remained a small local one. In addition, local training places have been started to provide direct hands-on training of local staff

[4]TACIS: Technical Assistance for the Commonwealth of Independent States (CIS): created by the EC upon request of the Member States as follow-up of the decisions of the 1992 G7 Summit in Munich to address nuclear safety and security issues in the CIS, which were of increased importance with the breakdown of the Soviet Union.

at some prototype nuclear installations, such as at the Ulba Fuel Fabrication Plant in Kazakhstan. [13]

Nowadays with joint efforts of EC and US/DoE the RMTC represents an internationally recognized training centre for nuclear safeguards, attended by inspectors of Rosatom and Rostechnazdor, staff of nuclear installations in CIS and even IAEA inspectors. Yearly 5 weeks of basic nuclear courses, 5 weeks of nuclear materials control courses, 7 weeks of NDA courses, 4 weeks on statistical methods courses, 5 weeks of software engineering courses, and 4 weeks of inspection course are offered in Obninsk and RMTC organized already 6 international seminars. Initially the lecturers at RMTC were assisted by JRC experts, but are now in a sustainable manner managing the training independently in a sustainable manner.

4.3. New Training Initiatives for Nuclear Security

End 2005 a Working Group dedicated to Border Monitoring (BMWG) has been created under the auspices of the IAEA, with the US DoE/NNSA[5], JRC, DG RELEX, DG AIDCO and the EU Council as members, to coordinate internationally the activities in implementing the support programmes against illicit trafficking of nuclear and radioactive materials (N/RM). Under the Second Line of Defense (SLD) programme joint training actions, at national and regional levels, are organized for efficient N/RM detection at borders while available resources and means are shared between the three main participating organizations (IAEA, NNSA and JRC). [14] A common syllabus has been finalised to further provide an integrated approach in the field. The US DoE/NNSA proposed a worldwide integrated training approach in collaboration with IAEA and EU, sharing information and experience of two one-week courses (i) the International and Domestic Border Security Training (Interdict/RADACAD) conducted at PNNL's HAMMER[6] facility and (ii) the On-site Radiation Portal Monitoring and Handheld Equipment (in-country) Training, which fits within the Megaports Initiative. [15]

The collaboration between NNSA and IPSC is extending to the support for the creation of a EU nuclear Security Training Centre (SeTraC) at Ispra to provide training courses related to the detection of N/RM and corresponding response on the SLD/HAMMER model. IPSC will then give hands-on training to front-line officers, dealing with response to detection of N/RM at borders (typically equipped with portal monitors) and other relevant nodal points.

This training is addressed to front-line officers of EU27, but also of countries that benefit from the EC support under the instrument of stability (such as some of the Commonwealth of Independent States, Lebanon, etc.). It can even be envisaged in the future to include in this type of training, issues of dual-use goods and import/export control (if the collaboration with US-DOE will also continue in this area).

The training centre will be used by both partner organisations (i.e. SLD and IAEA) for their own needs. The pilot session is planned in September 2009.

5. Conclusion

Based on the lessons learnt from the past, an approach to educate and train the next generation safeguards specialists in a continuous way with integrated effort is proposed. Synergy between international education and training experts allows for better student outreach, for better utilization of existing education programs, and for the involvement of industry along side government and Non-Governmental Organizations in the development of safeguards education and training initiatives.

The ESARDA course on Nuclear Safeguards and Non-Proliferation gives an example of a multidisciplinary course, addressing as well as technical as juridical and political issues. It provides an asset to the academic curricula of both the nuclear engineering faculty and the

[5] NNSA: US Department of Energy's National Nuclear Security Administration

[6] HAMMER: the Hazardous Materials Management Emergency Response facility at the Pacific Northwest National Laboratory (Richland, Washington)

nuclear law faculty by the expertise of the teachers, the unique safeguards laboratories and the open discussions in the exercises, all provided under the premises of a European neutral research centre. With its high demand the repetition more than on annual basis and extension beyond ESARDA participation is envisaged.
In order to further benefit from synergy between organisations teaching nuclear safeguards, non-proliferation and nuclear security, an international working group is established under the name NUSASET (Nuclear Safeguards and Security Education and Training). Description of its scope/content is outlined in the paper and annex.

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SESSION 21

FACILITY SPECIFIC SAFEGUARDS

Laser-based Monitoring of UF6 cylinders

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Abstract:

Development efforts are in progress to enhance safeguards implementation at uranium enrichment facilities. One component of the enhanced approach is the systematic tracking of UF6 cylinders between process and storage areas as well as between different process areas. The main challenges are the means of tracking the cylinders reliably with a minimum of operator involvement, and finding the optimum location to install a monitoring system which is capable of recognising all types of cylinders as well detecting non-standard containers.

A Laser Item Identification System (L2IS) has been developed successfully that is capable of monitoring all transfers of UF6 cylinders between process areas. L2IS uniquely identifies each cylinder through exploring the unique microstructure of each cylinder's surface. It has been demonstrated that every cylinder has a unique 'fingerprint' which remains intact even under extreme environmental conditions. The L2IS system is composed of a portable unit, operated in attended mode, and a fixed installed unit, operated without inspector presence. The portable unit acquires the fingerprints of a given set of feed cylinders intended to be used over the coming months and the fixed system monitors the flow of previously identified cylinders in a transfer corridor. This system is coupled with standard video surveillance that can remotely transmit state of health information to IAEA Headquarters. The video surveillance can be interfaced with electronic seals applied to the cylinders to record and display seals data (e.g. status, time/date of application). The integration of data from the L2IS with data from weighing and NDA stations is foreseen to monitor and verify all transfers. This will provide a high deterrence of diversion or substitution, and an increased probability of detection thereof. The paper will describe the successful results of the L2IS after a year of field testing, the intended use of the L2IS during inspection, and the possible additional integration of other monitoring capabilities.

Keywords: Monitoring; Enrichment; Integrated System; UF6 cylinder

Introduction

The IAEA closely monitors nuclear activities and material – particularly when fuel cycle activities could yield material readily available for weaponization such as highly enriched uranium (HEU). Therefore, the application of robust safeguards measures at nuclear facilities enriching/depleting and processing uranium is of vital importance to the IAEA.

The large size, complexity and increasing automation within the process areas of enrichment plants has resulted in intensive efforts to identify measures to enhance safeguards approaches. The IAEA is studying alternatives to optimize its use of inspection resources while at the same time maintaining credible safeguards implementation. The potential of an unattended approach to effectively safeguard quantities of depleted, natural and enriched uranium during the interim period between inventory takings has been considered as a consequence of affordable verification solutions to perform the monitoring of cascade areas not yet having been identified.

Resources will be used more effectively and efficiently by verifying items at entrance and exit points, which minimizes the use of expensive instrumentation within process areas and reduces the possibility of failure or incomplete coverage. Clearly, a 'hands off' approach is also favoured by the operator, as it provides for less intrusion into the process environment. This inspired the development of a novel

instrument capable of identifying positively and monitoring items both entering and leaving the process areas of bulk handling facilities.

System Requirements

Unique identification of items as part of a verification system is vital in providing real-time information on the location of such extremely sensitive and valuable material as UF6. To fulfil its safeguards purpose, the system must also handle the complete range of standardized transportable containers, including internationally transported UF6 cylinders (e.g. natural uranium (NU) feed in type 48Ys, NU heels in type 48Ys, or low enriched uranium (LEU) in type 30Bs) as well as national and facility-specific cylinders. Attempts to use non-standardized containers should also be detected and recorded.

Surveillance-based Solution

Standard surveillance alone is insufficient for monitoring the movement of cylinders for safeguards purposes. The comparison of two-dimensional images and Optical Character Recognition (OCR) of the ID plate from the cylinder were evaluated and rejected as they may be readily defeated. The two-dimensional "identity" of the cylinder cannot be uniquely maintained as the ID plate may be exchanged (albeit with difficulty) between cylinders. Depending upon cylinder type, some identification plates are screwed or riveted in place, while most are welded. ID plates can be re-engraved, e.g. during a quality assurance operation (operator dependant). Variations in shape, material, surfaces and text layout (manufacturer dependant) of the ID plate also pose technical difficulties in reading and identifying the plate.

Identification Tags

In 2005 a project aimed at developing a system for monitoring cylinders studied the possibility of applying tags to cylinders in order to positively and repeatedly identify the cylinders. The application of tags must however be performed by inspectors in an efficient manner (in less than 2 minutes), without involving welding (due to safety requirements), and still guarantee that the tag will not become detached during operational processes (e.g. heating, cooling, transportation, lifting etc.). Consideration was also given to whether a tag might be applied on the side surface by gluing, or by using one of the cylinder's skirt holes (using wire). In order to meet basic requirements, these tags would need to be:

- Readily authenticable;
- Tamper-resistant and tamper-indicating;
- Resistant to process environment: autoclave, cold traps, outdoor/indoor storage, etc.;
- Low cost (important given the number of cylinders involved).

During the project an evaluation of the likely effort required for 'tagging' was a major factor leading to the search for an alternative solution. Verification activities needed to be reasonably economical and time-efficient, particularly considering the potentially large number of cylinders involved. Applying any additional engraved tags/labels, which would lend themselves to OCR, was eventually entirely dismissed; in short, the need to weld tags to cylinders was unacceptable to operators[1] and too time-consuming for IAEA inspectors.

A solution based on passive radio frequency tags (RFIDs) was also considered but rejected based on the difficulties linked to their attachment to a cylinder in a reliable manner that would in turn guarantee authentication. In this case glue, weld or belts (several solutions were evaluated) did not fulfil the criteria of positive attachment and identification, capability to remain intact throughout the process handling, and cost-effectiveness.

[1] It is important to underline that any direct welding based solution was immediately dismissed by operators because this has safety and licensing/testing implications.

Reflective Tags and Reflective Particle Tags

Also evaluated were reflective tags of the type that could be randomly cut into different sizes and shapes and then applied to the surface of a cylinder (the pattern thus formed being read by a laser scanner and recorded as the unique identity of a cylinder). Again, no tags were identified that were sufficiently robust to survive process handling and this solution was therefore rejected. The alternative of replacing the reflecting tags by reflective particles was also studied. The reflective particles would be incorporated in a kind of gel that could be applied to the cylinders, producing a random pattern. The challenge of this technique consisted of finding a gel both sufficiently translucent to enable visibility of the reflective particles and resistant to process conditions (temperature, radioactivity, scratches), which could alter the transparency of the gel and affect the reading of the pattern. The gel needed to be sufficiently user-friendly to be accepted by an operator and capable of being applied expediently and reliably by inspectors. Although several demonstrations were performed, operators were in principle resistant to the application of any kind of chemical paint or glue to cylinders.

Intrinsic solution

Since no gel system or tagging solution appeared cost effective or met with technical or operator requirements, the need emerged to use the cylinder's own structure as an intrinsic identification (similarly to a fingerprint). Nowadays, fingerprinting has many industrial applications, resulting from security concerns (access to buildings, online payments, counterfeiting of passports, banknotes, pharmaceutical products, etc.). The known presence of fingerprinting can both deter and combat deception. Taking the fingerprint of a UF6 cylinder was not an application considered previously by laboratories or research institutes, but was pursued by the IAEA.

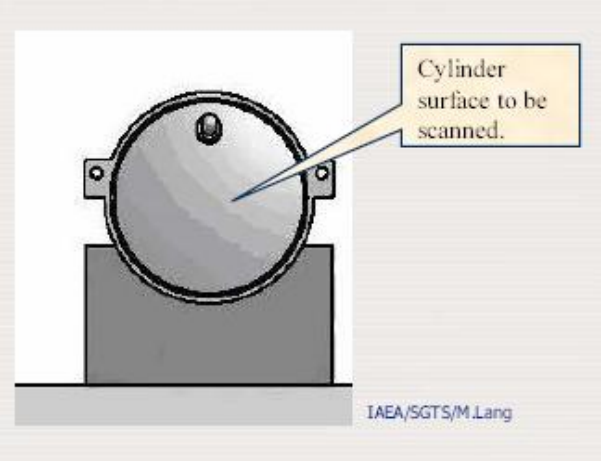


Figure 1: UF6 cylinder side surface to be scanned

Feasibility Study of Laser- based Systems

In 2006, feasibility studies of two types of 3-D laser scanning systems for fingerprinting were evaluated in parallel: the Laser Surface Authentication (LSA) technology from INGENIA Ltd (UK) and an approach from the EC Joint Research Center from Ispra (Italy) entitled 3-D Laser Surface Mapping (3D-LSM). Both techniques were evaluated with a view to the same application: scanning a mock-up of a UF6 cylinder type 30B, and being able to identify it uniquely. The two technologies were tested with four surfaces on two cylinders: one dully painted and one unpainted (representing two extremely different surface characteristics) and read at a range of less than one metre.

The study revealed that the LSA technique is not suitable for use on either a painted or unpainted drum surface, due to both surfaces having highly unusual optical properties. The only viable option for the use of LSA would have been the addition of a plate to the drum. The addition of a plate was deemed not to be possible (see above) and therefore the LSA was discounted for this application. The LSA evaluation was nevertheless a useful exercise for the IAEA, which identified a cost efficient application for this technology in the containment area [See LSA use for Metal seal verification-[1]-].

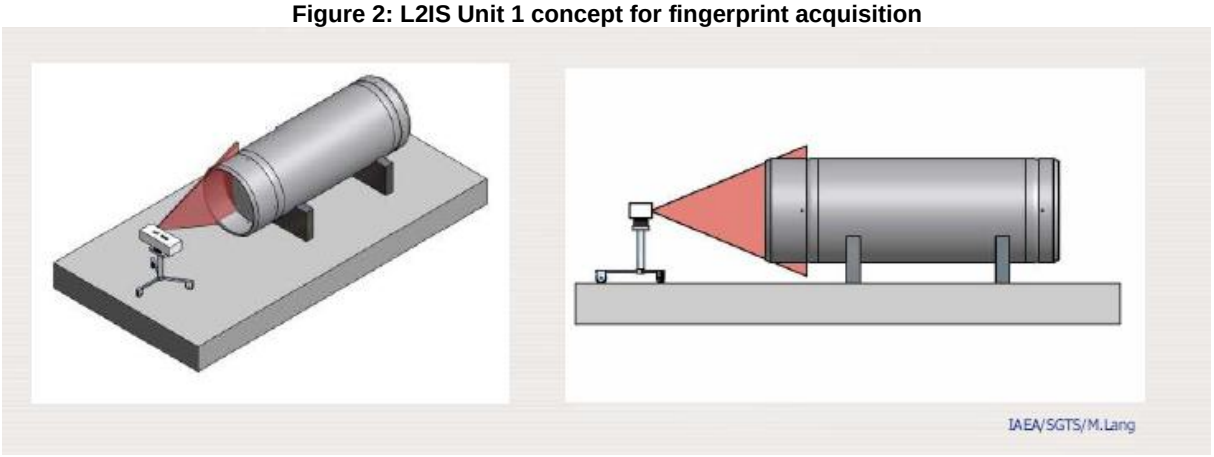
The testing of the 3D-LSM both with dully painted as well as shiny surfaces yielded good results. This technology provided a unique identification and should be robust enough to cope with a large variety of cylinder surfaces [2]. The 3D-LSM technology was then to be adopted by the IAEA and a Laser Item Identification System (L2IS) based on this technology was installed in October 2006 for field testing.

Laser Item Identification System Configuration

The verification of the unique identity by the L2IS is achieved by comparing reference items, which are recorded in a database, with the actual items. The system is therefore composed of two subsystems:

- A referencing unit (Unit 1) which acquires the fingerprint (reference scanning), and
- A verification unit to authenticate the fingerprint (verification scanning) of cylinders passing by (called Unit 2). The verification unit must be located in a well defined 'key position' in order to monitor the cylinder movements.

Figure 2: L2IS Unit 1 concept for fingerprint acquisition



The system is coupled with standard surveillance to ensure that no cylinders or other containers could pass by unseen. The cylinders are transported from one side of the hall to the other side on rails — the location of the system benefits from this unique transit path between the process and storage area.

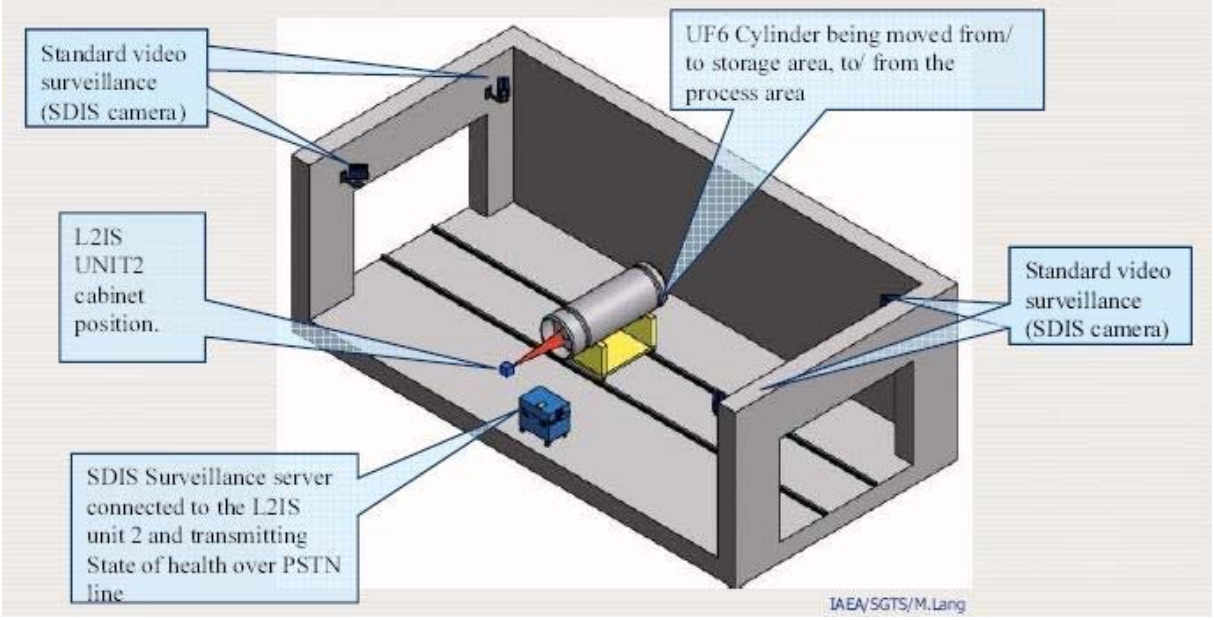


Figure 3: L2IS unit 2 concept for unattended identification coupled with surveillance (SDIS)

Both systems (surveillance and laser based scanner) are synchronized — the data is reviewed in a unique review interface, and the latest General Advanced Review Software (GARS) is used to jointly review both video and 'cylinder identity' data.

The L2IS project uses reliable, mainly standardized and commercially available, components and is implemented in a standard IAEA cabinet, offering to the inspectors standard laptop data retrieval media and procedures, which minimizes the need for additional training and maintenance.

In order to complete the equipment system, a third unit will be considered for in-situ verification during physical inventory verification (PIV), to enable random verification of (static) stored cylinders. The development of this third unit will commence in the second half of 2009, now that the fundamental technology has proven to be successful both from a methodological and implementation perspective.

Technical Description

The function of the 3D-LSM is based on laser triangulation (illustrated in Figure 4 below). A sheet of light is projected onto the interrogated object using a laser diode and a cylindrical lens. Where this sheet of light intersects with the object surface, it creates a laser line which is viewed by a digital camera from a different angle. The camera is equipped with a bandpass filter adapted to the laser wavelength so as to minimize the influence of ambient light. The shape of the laser line 'seen' by the camera depends upon the shape of the object. Assuming that the system is properly calibrated, each point of the laser line recorded by the camera yields the coordinates of one point in the laser plane. Each camera image therefore yields a profile which is the intersection between the laser plane and the object surface. By moving the object or the scanner in a controlled manner, a sequence of profiles can be acquired, producing a dense cloud of 3D points on the object surface.

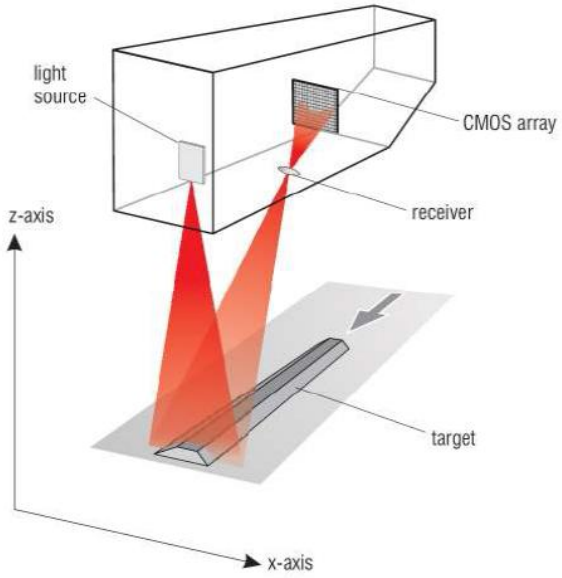


Figure 4: Triangulation principle (figure courtesy of Micro-Epsilon)

Project Testing and Technology Evaluation

The first part of the evaluation of the technique was aimed at selecting the range and model of laser scanner requirements for the foreseen operating conditions. This was achieved during a first field test, where a wide variety of static cylinders were scanned using laser scanners with different energy levels at BNFL Springfields. (December 2006).

Once the system requirements and needs were established for the unit in charge of the reference acquisition, it was tested on a second set of cylinders in a different facility, under a support task from the French Support Programme. A test in Pierrelatte (AREVA) had two further objectives:

- Determining the requirements for scanning cylinders during transport (cylinders in motion), and

- Conducting an aging study of cylinders in order to determine if the standard cylinder process handling and maintenance cycle would affect the surface structure.

Figure 5: L2IS unit 1 prototype (2007) (Pict. EC/ JRC Ispra/ISPC)

The first scans were acquired in July 2007 and the second campaign took place in May 2008.

The results of these tests identified the need for further development of the verification unit to be capable of scanning cylinders during transport. The speed factor is indeed a major influencing parameter but the occlusion/shadow created by the structure of the cylinder skirt and the bars on the trolley in regard to the triangulation scanning device was also a significant challenge to overcome. These were important points since the scanning system is not symmetric and the use of the system was intended to be able to scan a transported cylinder in both directions (from right to left and left to right).

In September 2007, the first trial L2IS system was installed under actual conditions, aiming to monitor a single cylinder (30B type) (one geometry) while being transported between areas in a facility. The system was used when the cylinder was in a static condition:

- The operator stopped the transport trolley for a few seconds in front of the verification scanner (UNIT 2);
- The laser scanner was mounted on a rotation axe, performing a semi-circular swipe and permitting the laser beam to sweep across the cylinder surface.

The acquired scan was then processed and compared with the reference database. The reference database was populated with scans obtained by use of the reference acquisition unit, operated by an IAEA inspector. The Operator had to provide access, for reference acquisition, to all cylinders that were expected to be used over a material balance period.

12 Month Trial Results

After a 12 month trial, during which the verification of a significant number of cylinders (about a hundred) was performed, the L2IS Unit 2 Version 1 prototype provided data that allowed for verification of the operator's cylinder movement declarations.

Thanks to the strong participation of the operator in the L2IS project, the IAEA was able to acquire reference scans (fingerprints) of most 30B type cylinders intended to be used during the 12 month period. The following table depicts the results of the trial. Note that, if some cylinders were presented for verification to the unattended unit 2, but had not been made available for referencing beforehand, no identification was possible. A later reference acquisition could however be made, the reference database incremented and the verification scan (for which there was initially no match re-compared with the updated database). In this case the comparison would not be automated but manually triggered (i.e. launched by the inspector using the L2IS automatic comparison software). Therefore the first 14% (corresponding to 16 cylinders for which there were initially no reference match in the database) could later be successfully verified.

GLOBAL RESULTS	
Number of cylinders referenced in the database and presented for verification (possible pairs)	**95 cylinders**
Number of positive identification: presented cylinders for verification that had previously been referenced	**100%**
Number of cylinders presented for verification and that had not been referenced (plant process constraints)	**14% (note that in the meantime all cylinders were referenced and are now present in the database)**
Number of non-matches (for movement where a reference was in the database)	**0**
Number of false matches (declared movement in regard to verified scan)	**0**
Number of missed cylinder movements in comparison with Operator declaration	**0**

Table 1: 12 month trial of L2IS Unit 2 version 1

The unattended system was operated by the facility staff after training by the project developers (IAEA+ EC) to acquire verification scans of all transported 30B cylinders. The operator's interaction was limited to the following two steps:

- Stopping the trolley transporting the UF6 cylinder in front of the L2IS Unit 2 Version 1,
- Activating the scanner via a key and button.

Simultaneously, the standard surveillance (DCM14 based SDIS camera) data were acquired, providing assurance that all transported 30B cylinders had indeed been scanned. These two operator interventions were part of the first prototype testing — the next generation aimed to release the operator from the need to stop and to activate the scanner.[2]

The early data obtained from this first generation system were analysed by the IPSC team, comparing different data processing options for the cylinder scan in order to find the optimal mathematical computing algorithm to apply in maximizing the L2IS system's capabilities.

This led to the adoption of a new procedure for the scan acquisition by Unit 1 (double swipe of the surface) to cope with frequent cases of the paper label pealing off the cylinders (on a majority of the cylinders presented during the first trial the artefact introduced by the peal-off label presented an interesting challenge and, in response, a reliable and uncomplicated solution — from a procedural standpoint — had to be found). This solution was implemented on the second generation prototype of Unit 1 (involving software changes only) and was tested in March 2009. While the data generated from these latter tests were being analysed, the system originally classified as Class 3B (class of the discreet components) was re-authorized and certified as Class 3R.

Current System

An upgraded version of L2IS has recently been installed at an enrichment plant and is expected to run for an initial further trial period of at least 6 months, scanning three different types of cylinders during this time period.

[2] Key activation is linked to the requirements from the IEC 60 825.

The latest system operates in a fully unattended and automatic manner. The system has been integrated with the SDIS surveillance system in a unique cabinet, aiming at providing a unique and standard servicing interface and procedure.

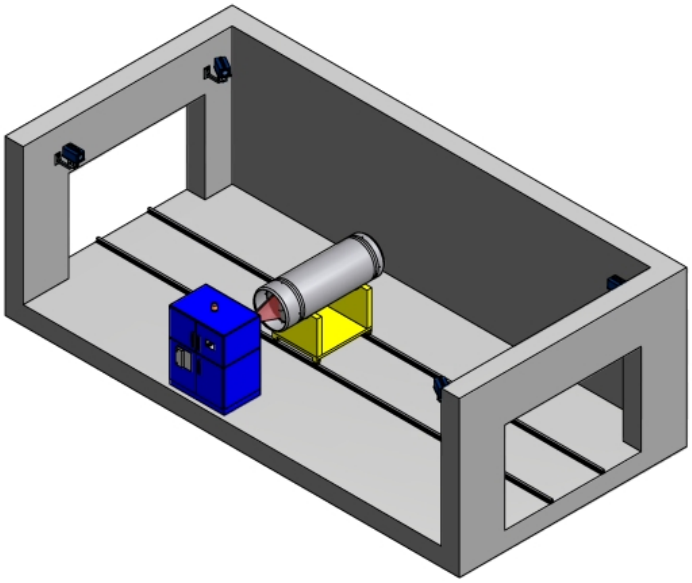


Figure 6: L2IS Unit 2 Version 2 as installed in March 2009

To be able to identify the three types of cylinders existing in the facility in question, a wide scanning campaign was organized with the operator in March 2009, which permitted the scanning of all 30B, 48Y and intermediate product cylinders intended for use in the coming months. The database of the installed L2IS Unit 2 Version 2 was then populated with this data and considered ready to verify all cylinders that would be presented as part of the theoretical 'Operator declaration'.

The second prototype of the L2IS Unit 2 was developed with the aim of achieving:

a) A fully automated identification system (i.e. no intervention from the plant operator required), and
b) The capability to identify the three type of cylinders used in the enrichment facility in question (different types imply different diameters and associated scanning configurations).

Future Expansion of L2IS

Within the toolbox of technical measures applicable for safeguards implementation at enrichment facilities, the capabilities of the L2IS could potentially be further expanded to enable the characterization and quantification of nuclear material contained in the monitored cylinders. Such a combined, comprehensive system could employ laser identification, weighing, and determination of Uranium mass and enrichment while under surveillance.

Furthermore an advanced L2IS system could act as a 'barrier' on the transit route between process and external storage areas, being located at the entrance/exit point(s). A transfer trolley in the process area could be stopped and the cylinder lifted onto a modular scale/scanner/NDA unit, before its onward transfer to the storage area.

Additionally, all NU feed cylinders could be scanned in advance at point of origin (when filled at the UF6 manufacturers), as could the empty 30B containers (at the cylinder manufacturers). The data generated could then be transferred electronically to the IAEA, allowing for the automatic identification of the cylinders at any location where a laser tracking device had been installed.

Conclusion

The challenge to uniquely identify the flow of various types of UF6 cylinders has been successfully addressed through the joint development and testing efforts of the IAEA and JRC. A novel instrument has been developed, which is capable of positively and uniquely identifying and monitoring items containing UF6 both entering and leaving the process and storage areas of bulk handling facilities.

Successful verification depends on the completeness of the reference database. The success of the trial described herein was heavily linked to the cooperation and support of both the operator and the State system of accounting for and control of nuclear material (SSAC).

Assisted by the European and French Support Programmes to IAEA Safeguards, the L2IS project has advanced within a relatively short time frame. In-situ testing is now underway in preparation for field implementation. The L2IS system provides the IAEA with an automated system to uniquely identify and thus monitor the complete flow of UF6 cylinders in enrichment plants in an effective and efficient manner. The system is relatively non-intrusive and to a large extent automated. In its present form, L2IS presents a building block to the comprehensive application of safeguards measures to nuclear material bulk handling facilities and, in particular, to enrichment plants. Pursuant to the equipment toolbox approach towards providing suitable and adequate equipment for safeguarding enrichment facilities it is planned that the L2IS system will be complemented with additional systems capable of monitoring and quantifying the nuclear material contained within UF6 cylinders. As a key component of modern safeguards implementation at enrichment plants, L2IS is expected to reduce in-field IAEA inspection resources while at the same time maintaining credible safeguards implementation. L2IS lends itself readily to the concept of remote safeguards inspections, whereby safeguards data are acquired from the field without the physical presence of an inspector.

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Geological Repository Safeguards: From Development to Implementation

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Abstract

During the past 20 years, the International Atomic Energy Agency (IAEA) has managed a number of Member State Support Program (MSSP) projects to support the development of safeguards for spent fuel conditioning plants and geological repositories, including the Programmes for the Development of Safeguards for the Final Disposal of Spent Fuel in Geological Repositories (SAGOR and SAGOR II) and the Experts Group on the Application of Safeguards to Geological Repositories (ASTOR). At the ASTOR meeting, held in September 2008, the IAEA presented to the Experts Group draft model integrated safeguards approaches and design information questionnaires for spent fuel assembly conditioning plants and geological repositories and received and resolved the comments of the participating State experts. These draft safeguards approaches were based on the safeguards approach recommendations made through the MSSP tasks and IAEA safeguards policy and guidance regarding integrated safeguards implementation. The model safeguards approaches will, when approved, provide guidance for preparing a safeguards approach for a spent fuel assembly conditioning plant or geological repository in a State under integrated safeguards. In December 2008, discussions on safeguards implementation for the planned geological repository systems in Finland and Sweden were held with the European Commission, Finland, and Sweden. Technologies and procedures necessary for verifying the repository design information are also being investigated. This paper summarizes the safeguards approach development efforts, describes the draft model integrated safeguards approaches for spent fuel assembly conditioning plants and geological repositories, and reports on the status of preparations for safeguards implementation at the repository in Finland.

Keywords: safeguards; geological repository; spent fuel; IAEA; Finland

1. Background

Approximately 20 years ago, in September 1988, the International Atomic Energy Agency (IAEA) hosted the first advisory group meeting addressing safeguards for the final disposal of spent fuel and wastes. An outcome of this meeting was a series of consultants meetings whose recommendations resulted in IAEA policy papers on termination of safeguards in measured discards and on safeguards for the final disposal of spent fuel in geological repositories. Another outcome of the meeting was establishment of the multi-member State support program (MSSP) task on the Programme for Developing Safeguards Approaches for the Final Disposal of Spent Fuel in Geological Repositories (SAGOR) in 1994. The SAGOR Programme final report[1] was published as a 4-volume report containing the Executive Summary (Volume I) and model safeguards approaches for spent fuel assembly conditioning facilities, operating repositories, and closed repositories (Volumes II-IV). A 1997 Advisory Group Meeting reviewed the safeguards studies and recommended use of the model safeguards approaches by the Agency.[2] The multi-MSSP SAGOR II Experts Group was initiated in 1999 to evaluate and make recommendations on the verification measures and technologies proposed for use in the model safeguards approaches and to coordinate development efforts to prepare the IAEA for implementing the safeguards approaches. SAGOR II held eight experts group meetings and produced reports on "Use of Geophysical Techniques for Safeguarding Geological Repositories"[3] and "Interface Issues and Interaction between Safeguards and Radioactive Waste Management in the Context of Geological Repositories".[4] The multi-MSSP ASTOR Experts Group was

initiated in 2004 to advise the IAEA on the application of the model safeguards approaches and technical measures to specific repository systems. The SAGOR, SAGOR II, and ASTOR programmes have had the active involvement of six to ten MSSPs, as well as the IAEA and Euratom.

In parallel to the SAGOR, SAGOR II and ASTOR programmes, additional MSSP tasks, as well as national development and evaluation projects, have been conducted on monitoring technologies that could support the implementation of safeguards for the geological repository systems. These development tasks have included verifying spent fuel assemblies, canisters and casks; geophysical monitoring; 3-dimensional (3-D) laser modeling; and satellite surveillance. Finnish regulatory and operational activities monitoring the construction of the Onkalo underground characterization project have proven the applicability of passive seismic arrays and 3-D laser modeling.

From the first IAEA meeting, the involvement of geological repository system operations and safety experts have been involved in the process of developing the model safeguards approaches and measures for the geological repository system. One of the SAGOR II meetings specifically addressed the interface between IAEA safeguards, facility operations, nuclear safety, and nuclear security, including the potential for joint use of monitoring systems.[4] The safeguards experts also cooperated with the IAEA Department of Nuclear Energy in developing a Nuclear Energy series technical document addressing the implications of IAEA safeguards on geological repository operations.[5]

2. Current status

In 2001, Finland's Parliament endorsed the construction in Finland of a geological repository system. In 2003, Finland provided to the European Commission and the IAEA information regarding its repository program, including its plan and foreseen activities for constructing a geological repository at Olkiluoto. This declaration stated that the Onkalo project would perform the underground characterization work. Upon completion of the Onkalo project, if the location meets the licensing requirements, the construction of the Olkiluoto repository will be initiated. Construction began in 2004 on the ramps and shafts that are the access points of the Onkalo project, and will become the access points to the Olkiluoto repository facility, if approved. As of the end of 2008, the entrance ramp had reached a depth of more than 300 m. In 2006, the Swedish Nuclear Fuel and Waste Management Company (SKB) submitted license application information for its spent fuel conditioning plant to what is now the Swedish Radiation Safety Authority (SSM).

In early 2008, using the recommendations from the SAGOR, SAGOR II, and ASTOR programmes, the IAEA drafted model integrated safeguards approaches and design information questionnaires for spent fuel assembly conditioning plants and geological repositories. The draft integrated safeguards approach concept was reviewed the IAEA Director General's Standing Advisory Group on Safeguards Implementation (SAGSI) in May 2008, and the full documents were reviewed by ASTOR in September 2008, and by the Department of Safeguards in November 2008.

SAGSI advised that the Secretariat's policy paper on geological repository safeguards should still be considered relevant and important. With consideration of the status of construction of Finland's exploratory facility, SAGSI also advised that the Secretariat should take urgent action to agree upon the State-level approach for Finland and to develop the integrated safeguards approach and respective safeguards measures to be applied to the repository project.

During the November 2008 Lower Level Liaison Committee Meeting between the IAEA and European Commission, it was agreed to start formal discussions regarding the application of safeguards to Finland's proposed geological repository. Furthermore, the Commission agreed to request Finland to declare the facility declaration and provide design information for the repository. In December 2008, the IAEA held its first formal meeting with the European Commission, Finland, and Sweden to discuss repository and spent fuel assembly conditioning plant status, draft safeguards approach models, and actions necessary to implement safeguards for Finland's and Sweden's geological repository systems. The meeting identified initial actions that needed to be accomplished for the IAEA to begin verifying Finland's geological repository design. The European Commission is expected to formally declare Finland's Onkalo project to be a facility and formally submit design information for the facility in the near future. The IAEA expects to perform the first design information verification visit in mid-2009.

The IAEA has developed a safeguards implementation 'road map' that includes the following additional activities that need to be conducted to complete the design information verification and to prepare for verification of the receipt and disposal of spent fuel at the repository facility:

- Review the submitted design information for completeness and consistency;
- Draft a design information verification plan;
- Evaluate design information verification techniques for IAEA use at the Onkalo Project and Olkiluoto repository facility, including micro-seismic monitoring, 3-D laser scanning, ground penetrating radar, active seismic (localized), underground portable navigation systems, and satellite imagery (visual, radar, infrared, and/or multi-spectral);
- Perform periodic design information verification visits and complementary accesses;
- Develop a facility-specific integrated safeguards approach consistent with the State-level approach for Finland; and
- Develop and evaluate verification technologies for the operating phase of the repository.

The ASTOR Experts Group participants will meet in June 2009 to review the status of the geological repository systems, the IAEA's needed capabilities for implementing its verification plan, and the current status of development work on verification technologies related to geological repository systems. The expected outcome of the meeting is a program plan to initiate and complete the technical assessments and technology development needed to support the IAEA's verification activities.

The remainder of this paper will address the draft model integrated safeguards approaches developed for the spent fuel assembly conditioning plants and geological repositories. The facility-specific integrated safeguards approaches for Finland and Sweden will be consistent with the respective State-level approaches which take into consideration State-specific factors, safeguards implementation and evaluation results, acquisition paths, and the relevant safeguards technical objectives and performance objectives for the State and facility.

3. Draft model integrated safeguards approach for a spent fuel assembly conditioning plant

3.1. Assumptions

Within a geological repository system for the final disposal of spent fuel, spent fuel assembly conditioning plants are the last point at which spent fuel assemblies may be directly verified. The spent fuel assemblies will be received from at-reactor or away-from-reactor spent fuel storage installations that may be either wet or dry storage. If the conditioning facility is independent of the storage facility, the spent fuel assemblies are expected to be received in transport casks. The output of the conditioning facility is spent fuel assemblies permanently sealed in disposal canisters for direct emplacement in the repository. The spent fuel canister is expected to be removed from the facility in a shielded overpack that may be used only during transport to or into the repository or that may be a part of the disposal package.

The model conditioning plant is composed of buffer storage for received transport casks, a hot cell area for removal of the spent fuel assemblies from the transport casks, buffer storage for the spent fuel assemblies, hot cell area for repackaging of spent fuel assemblies into the disposal canisters, buffer storage for the disposal canisters and a hot cell area for loading the canisters into shielded overpacks. The model plant design assumes that spent fuel assemblies will not be consolidated.

3.2 Draft model safeguards approach

The draft integrated safeguards approach for the spent fuel assembly conditioning plant uses, as a benchmark, the IAEA policy paper on integrated safeguards for spent fuel transfer to dry storage. The policy paper requires that spent fuel assemblies must be verified by a partial defects test using the 'best available method' approved for inspection use, before becoming inaccessible for further verification. A spent fuel storage installation that packages fuel assemblies for dry storage and a spent fuel assembly conditioning plant perform the same general activities. However, while the primary

function of the storage facility is passive storage, the primary function of the conditioning plant is packaging large quantities of spent fuel for dry storage (also known as emplacement) in a geological repository. Thus, the integrated safeguards requirements for spent fuel storage and packaging for dry storage should also be applicable to the conditioning plant.

The design of the conditioning plant must be verified during construction and reverified during operation and decommissioning of the facility. During operation of the plant, design information will be reverified annually in association with the physical inventory verification. Because the function of the plant is to rebatch spent fuel assemblies between containers, the design of the plant is expected to be relatively simple and straight forward and the diversion paths will be limited. Design information verification is expected to be conducted mostly through visual verification, with use of other techniques as determined necessary.

Physical inventory verification also will be conducted annually. The spent fuel inventory will be verified in accordance with the requirements for verification of spent fuel in storage elsewhere in the State – spent fuel under successful dual containment and surveillance (C/S) will not be required to be verified and spent fuel under successful single C/S will be item counted. Spent fuel not under successful single C/S will be item counted, identified, and verified for gross defects with a random low detection probability.

When spent fuel is received, the tamper-indicating seal and/or other C/S systems on the cask will be verified, if present. The number of spent fuel assemblies and their identifications will be verified when the transport containers have been unloaded. In some cases, the spent fuel will not have been verified to the required level at the shipping reactor or spent fuel storage facility or will have been shipped under no or unsuccessful C/S. In other cases the spent fuel may have been verified by ‘best available method’ at the reactor or spent fuel storage facility and maintained under C/S. Because the timeliness verification goal for spent fuel under integrated safeguards is one year, and the spent fuel is expected to be measured during repackaging within the timeliness period, no verification measurements should be necessary on the received spent fuel. Depending on the transport cask receipt area and canister buffer storage operations, different levels of C/S and monitoring may be required to maintain continuity of knowledge on the spent fuel assemblies.

Before the spent fuel assemblies are loaded into the disposal canister, they must be identified and verified by partial defect tests using the best available measure approved for inspection use. If the assembly has been previously verified to this level and maintained under successful C/S, reverification is not required. The loading of each disposal canister would be verified using appropriate surveillance and monitoring techniques. After the disposal canister has been filled, the permanent closure or sealing of the canister and the canister identification should be verified. Attributes of the filled canister should be established to permit future confirmation that the contents of the canister have not been altered. These attributes may include weight, heat emanations, radiation emanations, and canister microstructure. The spent fuel assembly storage and disposal canister loading hot cell areas should be maintained under a C/S system that provides assurance of continuity of knowledge equivalent to a dual C/S system. Remote monitoring of the C/S and monitoring systems will be used whenever possible.

When the spent fuel canisters are to be transferred to the repository facility, the canister should be identified before being placed in the shielding overpack, the overpack identity verified, and a dual C/S system applied.

Random interim inspections will be conducted to verify nuclear material flow and declared operations at the spent fuel conditioning plant. Because of the high throughput of the plant – that is, possible loading and sealing of one disposal canister per working day – the facility operator will be requested to make mail box declarations stating the location, content, and status of each spent fuel item in the facility. Depending on State- and facility-specific factors, at least one short notice or unannounced random inspection would be conducted per year. Additional visits may be required for equipment checks and instrument maintenance. Complementary access will be scheduled in accordance with the IAEA’s annual implementation plan for the State.

4. Draft model integrated safeguards approach for a geological repository

4.1 Assumptions

The geological repository facility is assumed to be composed of above ground and underground areas. The above ground area will provide interim storage for received disposal canisters. The repository may be collocated with the spent fuel assembly conditioning plant; in which case, the output storage of the conditioning plant and the input storage of the repository facility may be merged. The repository site will be the area or areas of the surface delimited in the design information provided by the State on which the above ground structures of the repository facility are located. The site would include all areas where a component of the repository, including ventilation shafts, penetrates the surface. The isolation zone surrounding the repository in the geological formation forms the containment structure of the repository.

Access to the underground area of the repository facility will be by ramps and/or shafts. The design-basis flow of nuclear materials is only from the above ground areas to the underground areas of the repository. However, the design of most repositories include the possibility of having to bring a spent fuel disposal container back to the surface, and some repository statutes require that the spent fuel remain retrievable for a stated period of time.

The IAEA policy paper on geological repository safeguards requires that safeguards on the spent fuel be maintained during the operational period and after the repository has been backfilled. The nuclear material in the spent fuel does not qualify as being ‘practicably irrecoverable’ and safeguards may not be terminated on it. Nuclear material accountancy remains the fundamental safeguards measure for the above ground and underground areas of the repository facility. However, the presence of spent fuel assemblies in disposal canisters and of disposal canisters emplaced in the geological repository are not to be directly verified. Continuity of knowledge on the spent fuel content of the disposal canisters is to be maintained by C/S on the containers, and continuity of knowledge of the containers transferred to the underground areas is to be maintained by C/S on all safeguards-relevant access routes into the repository and monitoring to assure the continued integrity of the containment provided by the geological formation.

The model integrated safeguards approach for geological repositories addresses only the pre-operational and operational periods of a repository facility.

4.2 Draft model safeguards approach

Design information verification will be an important safeguards measure for confirming the stated design of the repository and detecting undeclared activities. The repository construction status will be constantly changing from the start of excavations through repository closure. During repository operations new tunnels will be constructed and filled tunnels will be backfilled. Because the exterior of the repository cannot be directly observed, verification of the integrity of the containment provided by the repository by visual and geophysical means will be required.

The State should provide information on the original undisturbed geological site, the excavation plans, and information on the proposed design of the repository structures in the underground area of the repository facility when a proposed repository location has been selected for underground characterization. Design information verification and monitoring by the IAEA should begin with the investigation program at the proposed repository site. Above ground design information verification activities are expected to be conducted annually in association with the physical inventory taking and would include observation of the construction of the buildings. Surface mapping using satellite imagery techniques is also expected to be used to establish a surface baseline and to detect existing activities in the repository environs that may need to be further investigated. Underground design information verification activities may include observation and use of surveying techniques (including 3-D laser modelling) to verify construction activities within the tunnels, passive geophysical monitoring to verify locations of excavation activities, and active geophysical monitoring to resolve potential discrepancies between declarations and observations. Inspector access to the underground area is expected to occur annually in association with the above ground design information verification and periodically to verify changes in the construction status.

Physical inventory verification will be conducted annually for the above ground and underground areas. In the above ground area, the spent fuel disposal canister inventory will be verified as follows: (1) spent fuel canisters under successful dual containment and surveillance (C/S) will not be required to be verified; (2) spent fuel canisters under successful single C/S will be item counted, identified and a unique canister attribute (e.g., weight, heat, radiation, or canister microstructure) will be verified; and (3) spent fuel not under successful single C/S will be verified in accordance with actions specified by the DDG-SG. For the underground area, the IAEA will evaluate all C/S and monitoring information to ensure that continuity of knowledge equivalent to that provided by dual C/S has been maintained.

Receipts of spent fuel disposal canisters will be verified by item counting, container identification, and verification of the C/S on the container. Transfers to the underground area will be verified by item counting, container identification, and confirmation of canister attributes. Removals of spent fuel containers from the underground area will be verified by item counting, container identification, and confirmation of container attributes. Other containers removed from the underground area will be monitored to ensure the absence of spent fuel. Remote monitoring of the C/S and monitoring systems will be used whenever possible.

Random interim inspections will be conducted to verify nuclear material flow and declared operations. Because of the high throughput of the above ground area installations – that is, a potential transfer underground of one disposal canister per working day – the facility operator will be requested to make mail box declarations stating the location and status of each spent fuel item above ground or being transferred underground. Depending on State- and facility-specific factors, at least one short notice or unannounced random inspection would be conducted per year. These could include access to the underground area. Additional visits may be required for equipment checks and instrument maintenance.

Complementary access will be conducted in accordance with IAEA's annual implementation plan for the State and at those locations outside the site for which there is a question or inconsistency. Complementary accesses will be conducted at any location on the geological repository site (including in the geological repository facility) and to any location outside the site that the IAEA considers might be functionally related to the geological repository facility.

5. Summary

In 2001, Finland's Parliament endorsed construction of a geological repository in Finland. Construction started in 2004 on the ramps and shafts that compose the Onkalo underground characterization project and that would become features of the proposed Olkiluoto geological repository, if approved. In 2006, SKB submitted license application information for its spent fuel conditioning plant to what is now SSM. Since 1994, the IAEA has been developing safeguards policy and model safeguards approaches for spent fuel assembly conditioning plants and geological repositories. In 2008, the IAEA prepared draft model integrated safeguards approaches and design information questionnaires and submitted them for review by SAGSI, ASTOR, and the IAEA Department of Safeguards. The draft model approaches will become effective after all comments have been resolved. The Department of Safeguards, following consultations with Finland, Sweden, and the European Commission, has prepared a draft safeguards implementation 'roadmap' for the application of safeguards to the pre-operational phase of Finland's repository. The IAEA, European Commission, Finland, Sweden, and the other ASTOR participants are coordinating to provide the IAEA with the necessary capabilities to effectively and efficiently implement safeguards for current and future geological repository systems for the final disposal of spent fuel.

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Quality Management in the OSL-RRP: Experience from three years of operation

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Abstract
This paper describes the various components of the Quality System for the On-Site Laboratory (OSL) at the Rokkasho Reprocessing Plant (RRP). As the Laboratory has been in active commissioning for only three years, the quality of the analytical results are reviewed on a daily basis and procedures are regularly updated according to the principle that the methods must be 'fit for purpose'. Calibration- and QC-sample documentation complement the documents on routine samples. Recently, OSL intensified its participation in external QC programs, such as EQRAIN and plans to extend these activities in its effort to comply with the ISO 9001 and 17025 standards. Improved methods (e.g. automated separations, spectra evaluation software), ongoing training of staff members, improved information management, exchange of experts, advanced instruments and tailor-made tools are developed, reflecting the highly dynamic nature of the work. Examples about processes and results are illustrated, without trying to cover all aspects of the quality management.

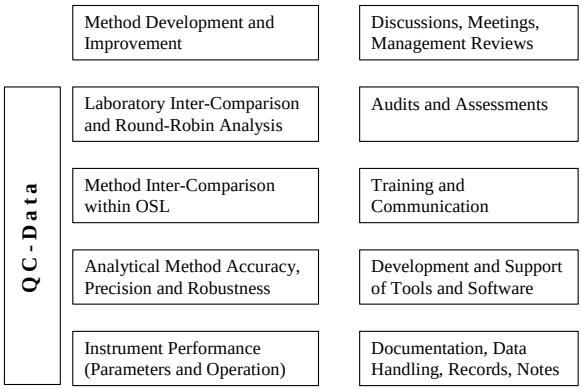
Key words: Nuclear Safeguards; Analysis; Quality Control; Quality Assurance; On-Site Laboratory

1. Introduction

As part of the safeguards (SG) approach to RRP, the OSL analyzes inspection samples originating from various flow streams and inventory points. The OSL analytical activities are shared between the Nuclear Material Control Center (NMCC) and the IAEA, because most of the equipment is installed as for joint-use. The Quality Control (QC) as one important component of the Quality System is also ensured jointly. QC procedures are being implemented for the various analytical methods (hybrid K-edge densitometry, X-ray fluorescence analysis, gamma spectrometry, isotope-dilution mass spectrometry, density measurements, spectrophotometry) and for the several sample treatment methods (dissolution, spiking, separation, weight controlling) for the characterisation of plutonium or uranium contents in samples of various types (dissolved spent fuel, product solutions, waste samples as well as MOX samples).

2. Quality System Overview

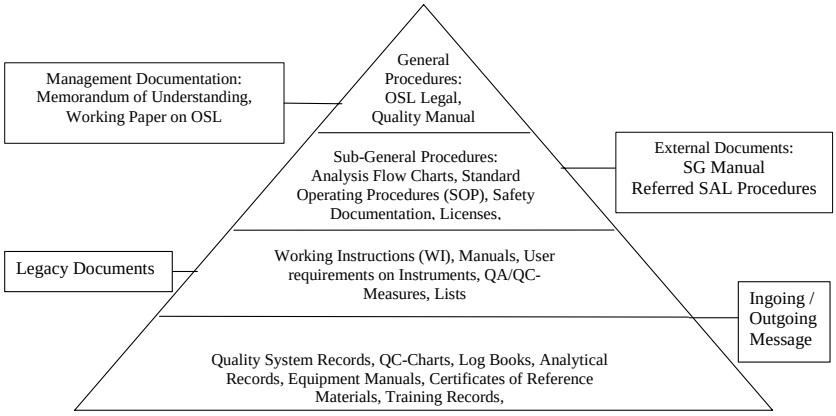
The various technical components of the OSL QS are summarized in Scheme 1. Bearing in mind that the OSL is at an early stage and that experience is building up with time, many of the analysis processes are dynamic and evaluative in their nature and aimed at further strengthening the QS by cross-linking processes, e.g. making them multiply redundant. Selected aspects of the components are discussed in the subsequent sections.



Scheme 1: Important components of the OSL-QS

3. Documentation

As a generality in terms of the Quality System (QS), the analytical core and auxiliary processes are documented at various levels, from general to specific, complemented by SG-specific procedures regarding authentication, continuity of knowledge and other aspects. Scheme 2 summarizes the documentation levels.



Scheme 2: Documentation system at OSL-RRP

The Quality System of the OSL which is still under development is documented in:

(i) The Quality Manual, which will give an overview of the Management System requirements as well as the technical requirements, referring to
(ii) a set of Standard Operating Procedures and Working Instructions containing detailed procedures and instructions relevant to working of the Quality System,
(iii) Joint use (IAEA and NMCC) documents, defined as 'OSL Joint Documents' such as instrument user requirements, manuals and QC procedures at the instrument level,
(iv) Quality System records (functional test reports, log books, records of analyses, certificates of reference materials, quality assurance records, quality standards, personnel records etc.), and associated documents. Comprehensive and up-to-date documentation is an important part of the Quality System, as outlined in the documentation policy. In that respect, the documentation structure (including the analytical data generated) on our server is currently reviewed for updates.

Three high-level joint documents constitute the basis for the operations at the OSL:

(i) The Memorandum of Understanding between the Ministry of Education, Culture, Sports, Science and Technology, the NMCC and the IAEA that is providing a legal binding document of the operations at the OSL,
(ii) the so called OSL-Working paper for the On-Site Laboratory at RRP that is gathering the principles, working arrangements and expected performances of the OSL, and
(iii) the Joint Notes of Operation that will specifically describe all interfaces between the IAEA and the NMCC for the operations at the OSL.

4. Examples in instrument and method performances

The basis of quality control rests on the instrument level: The QC data generated reflect the state of health of the various instruments, such as the short- and long-term stability. Indicative instrument QC data are confirmed to be within the limits, derived from the performance of similar instruments at SAL and other nuclear safeguards laboratories, before an analytical measurement is commenced.

4.1. Hybrid K-Edge Densitometry (HKED) and XRF analysis

In HKED for example, the count rates of the KED- and of the X-ray detector, measured from a control or QC sample, reflect the stability of the X-ray tube, the detectors and of the sample positioning. The correction factors derived from the intensity of the $U_{k\alpha 1}$ line (98.4 keV) in the XRF spectra of a QC–sample are used to normalize the XRF-result to the date of the latest calibration. Similarly, within a certain range corrections for the high voltage and for the temperature are applied in order to normalize the measurement results to the reference values (150.00 kV, 298 K). Other HKED instrument parameters under routine monitoring are the resolution of the 88 keV peak of the ^{109}Cd sources of KED- and XRF-detectors and the reference position of the sample changer.

On the next level of quality control at OSL, results from QC-samples and certified reference materials (CRMs) are checked against the reference values. For example, we implemented a QC-procedure for HKED as recommended by Institute for Transuranium Elements (ITU) of the European Commission Joint Research Centre to measure a certified synthetic sample containing U and Pu in the same batch with an inspection sample. Figure 1 shows the intensity of the $U_{K\alpha 1}$ line and the U/Pu-ratio of the QC-sample of one of our HKED-systems since the last calibration in February 2008. As the U/Pu ratio determination from XRF measurements in HKED mode is a relative method, it depends only indirectly on the count rate, which fluctuates significantly during the period shown in the Figure. The relative standard deviation of all U/Pu ratio results shown in the Figure 1 corresponds to 1.2%.

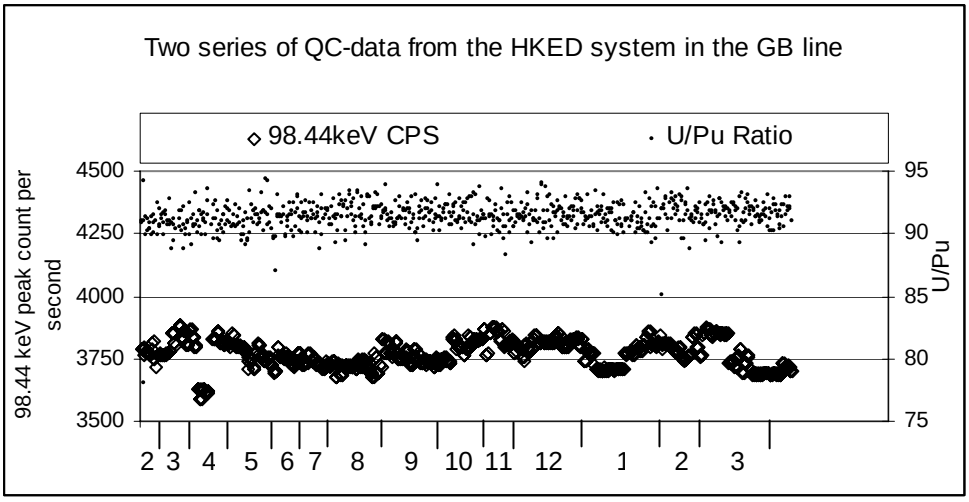


Figure 1: $U_{K\alpha 1}$ line count rate and U/Pu ratio (raw data, not normalized for changes in count rates) of a QC-sample containing 214.8 g/L U and 2.34 g/L Pu, measured in HKED mode for 15 months since the last calibration in the GB-line system. The shown count rate is not corrected to compensate for evaporation of the QC sample in use..

4.2. Random uncertainty in HKED

Among the references for the precision of the analytical methods at OSL are the International Target Values (ITVs), and specifically for the XRF measurements the ESARDA “Performance values for non-destructive assay (NDA) techniques applied for Safeguards”. In KED-U and KED-Pu at concentrations above 50 g/L, the random uncertainty component according to the ITV’s corresponds to 0.2%. This

compares with a standard deviation of 0.4% (KED-U) and 0.5% (KED-Pu) at both HKED-systems at OSL, derived from QC-samples and the calibration data, using data till June 2008. Since 2009, the X-ray tubes of the HKED analyzers were allowed to run continuously during the working week, which contributed to the reduction of the day-to-day variation of the HKED-results on QC- and inspection samples.

The target value for the random uncertainty of XRF-Pu on dissolver solutions samples (c_{Pu} 1 to 2 g/L) in hybrid mode (K-edge + XRF) is 0.6%. This compares with a standard deviation of 1.0% derived from calibration data and from QC-data since 2009. An alternative approach to determine the random uncertainty by taking into account the contributing sources (Figure 2) gave a comparable result (< 1.3% in 2008). The planned exchange of the X-ray tubes, which have now exceeded their guaranteed operation hours together with the introduction of new cryo – cycle detectors (Canberra) are expected to further improve the precision of the two systems installed at OSL. Also, the development in 2009 of new evaluation algorithms of the KED and XRF spectra in terms of a support program is expected to suppress time for measuring passive spectra and release time for the increasing of the count time in active X-ray mode that will contribute to the reduction of the random errors of the measurements.

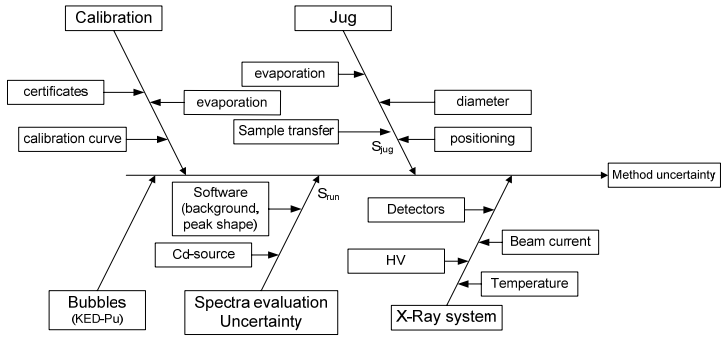


Figure 2: Sources of the random component of the method uncertainty

4.3. Thermal Ionization Mass Spectrometry

In thermal ionization mass spectrometry (TIMS) with total evaporation (TE) method, two filaments on each turret are routinely loaded with a CRM. The CRMs used for this instrument/method quality control are NBL-010 and NBL-137. NBL-144 is periodically employed to confirm the instrument stability over a wider range of isotope ratios and that the storage bottle with NBL-137 did not become contaminated. Blank filaments are measured periodically to check the cleanliness of the ion source.

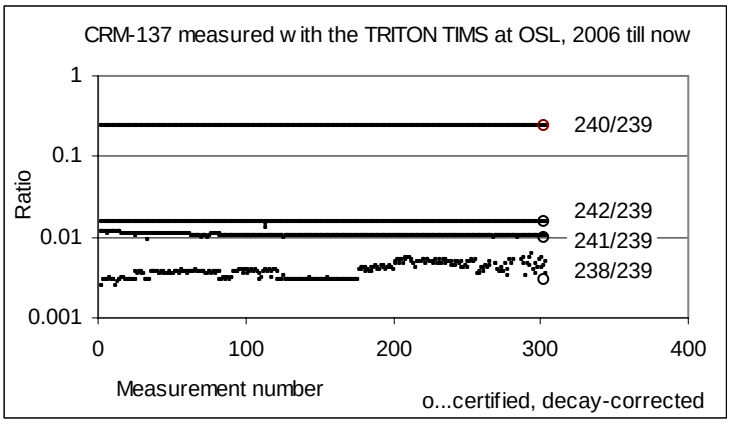


Figure 3: Isotope ratios of NBL CRM-137 measured by TIMS in TE mode at OSL

Figure 3 shows the isotope ratios of NBL-137, measured since 2006. Very few data are outside of the warning limits (WL) with causes which are traceable to technical reasons. The warning limits are: $WL_{NBL137:\ 240/239}$: ± 0.15% = 1 times the certified uncertainty, $WL_{NBL010:\ 235/238}$: ± 0.3% = 3 times the certified uncertainty. The decline of the ratio 241/239 in Figure 3 simply corresponds to the ^{241}Pu

decay. The circles at the end represent the isotope ratios expected from the decay-corrected certified values. The ratio 238/239 reflects the slight contamination of the ion source by uranium. The source cannot easily be baked out because, by design, the housing is attached to a stainless steel glove box frame. A first baking out (data point 124, Feb. 2008) with the built-in equipment gave nevertheless promising results. On the other hand, we cannot clean the ion source chemically due to the restrictions regarding waste generated at OSL.

5. Inter-comparison and round-robin exercises

5.1. Comparison between two methods

5.1.1 IDMS (Isotopic Dilution Mass Spectrometry) and HKED comparison

Figure 4 shows the relative difference between IDMS and HKED results as a function of time since the last HKED calibration, sorted according to inspection sample type. The latter is related to the composition and concentration range of the samples. Shown in this chart are KED-results (concentrations of U or Pu above 40 g/L); XRF-results such as IAT-Pu will be discussed separately in another paper due to the different concentration range.

On average, the deviation between [Pu KED – IDMS] accounts for – 0.3%, with a standard deviation of 0.9%. Thus, the deviation is statistically not significant. The average standard deviation was calculated in % according to $\sqrt{\frac{\sum (Dev_i - \frac{(\sum Dev_i)}{N})^2}{N}}$ with N being the number of data points (averages of triplicates) and Dev_i the relative deviation between HKED and IDMS. All these Pu-samples were measured in one and the same HKED-system (glove box, GB).
There are few results on KED-U in the first half of 2009, because no solutions from the dissolver were received at OSL. The average deviation [U KED – IDMS] in one of the systems (GB) amounts to – 0.05%, which is well-below the average standard deviation of 0.3%. In the other system (hot cell, HC), an average deviation of – 0.6% with an average standard deviation of 0.6% is observed for dissolver type solution-samples after applying correction factors for non-linearity. This system is now scheduled for re-calibration.

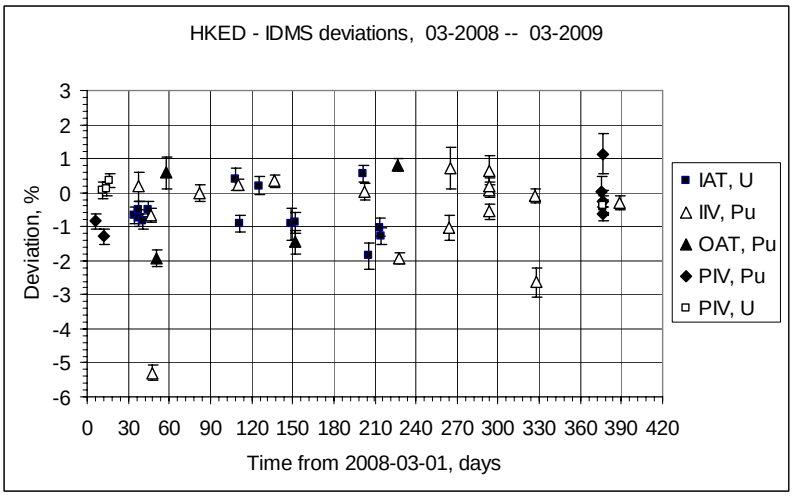


Figure 4: Differences between KED and IDMS results on concentrated samples ($c_{U\ or\ Pu}$ > 70 g/L) since the latest HKED calibration

5.1.2 Spectrophotometry of Pu(VI) and XRF

Within the calibrated concentration levels and in the absence of solid particles that might interfere with the sample preparation for the spectrophotometry of Pu(VI) , the differences between these two methods on low-level concentrations reach 10% at maximum. From this, a random uncertainty of the XRF method in standalone-mode is derived as 3% (sample type: diluted process solutions). The lower

limit of the currently calibrated range in XRF stand-alone mode at OSL is 0.5 g/L. The XRF uncertainty increases at low concentrations due to counting statistics and the low XRF to background counts ratio. Figure 5 shows that the relative difference between spectrophotometry and XRF results significantly increases to over 10% under the following conditions: (i) when the sample contains less than ca. 0.03 g/L Pu, and (ii) when the sample contains fines. It is concluded, that below 0.5 g/L Pu, spectrophotometry is more accurate than XRF, and that interfering particles ("fines") must be removed prior to spectrophotometry, e.g. by decantation as currently applied. A modified XRF-method is under consideration to account for the matrix in particle-containing samples.

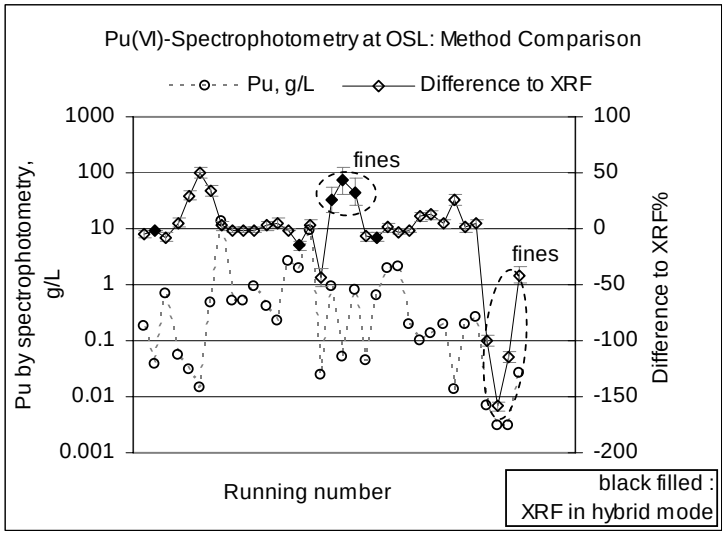


Figure 5: Differences between Pu(VI) spectrophotometry and XRF (standalone mode, except as indicated) for diluted samples

5.2. Data comparison between OSL and SAL on loaded filaments

As a first exercise on off-site analysis, ca. 80 filaments (40 duplicates), loaded with spiked (IDA) and un-spiked (ISO) separated U- and Pu-fractions of inspection samples, respectively, were shipped from OSL to the IAEA-SAL in Seibersdorf and measured by TIMS (MAT Finnigan 262).

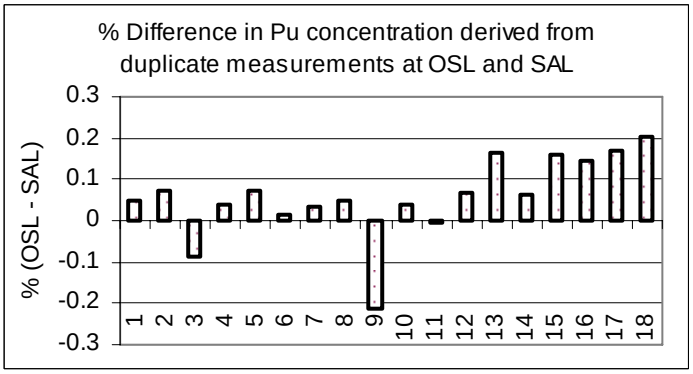


Figure 6: Comparison of results on loaded filaments after shipment from OSL to SAL

The IDMS-results as calculated from the raw data are compared with the results obtained at OSL on the same sample. Figure 6 shows the relative difference for 18 Pu-samples of different type. All results agree within 0.3%. The slightly positive average discrepancy originates from grown-in ^{241}Am as seen in the raw data on isotope ratios, because several factors caused a delay in the shipment. The sample #9 had a low signal intensity (filaments bent during transport), which contributes to the discrepancy. The test confirmed that off-site shipment is an option to confirm the quality of OSL mass spectrometry results, a back-up, and an additional authentication measure.

5.3 Participation in round-robin analysis

In 2008/2009, OSL participated for the first time in the EQRAIN-U exercise organised by CETAMA (France). The purpose for us was the control of two main measurement methods at OSL: IDMS and KED-U. Internally within the OSL, we compared results of the two methods with each other in order to confirm the stability of the KED components and to determine the extent of the KED-U non-linearity correction at that high concentration (which was outside of our calibrated range and outside of the usual inspection sample concentration). Additionally, we diluted the EQRAIN samples by a factor of ca. 1.7 in order to fall within the calibrated range of KED-U and statistically evaluated all results to derive the random component of the method uncertainty. At the same time, we compared KED-U results between the two systems installed in a glove box and in a hot cell, respectively. The largest absolute difference [OSL – certificate] was 0.05% (IDMS) and 0.17% (KED). These differences are smaller than the maximum uncertainties allowed according to the ITV's and fall within the method uncertainties at OSL, for IDMS even within the uncertainty of the certificate.
Overall, the participation in EQRAIN round 12 gave the OSL a good experience in treating external QC-samples and confirmed the approach implemented at OSL that IDMS serves as a accurate internal quality control of the HKED-systems, which are considered as the 'OSL-workhorses'.

5.4 Comparison with data provided by the operator

Since 2006, and on a monthly basis since June 2008, samples are selected by (IAEA)-OSL for which the operator of RRP is requested to analyze them according to a method specified by us and to provide the detailed analytical results after the final operator declaration (OPD). The results are then evaluated by OSL and discussed in the OSL technical meetings on a monthly basis. (As the OPD is already finalized, the results obtained with the specified methods cannot be changed by the operator). The specific purpose is to discover any systematic errors and deficiencies of new laboratory equipment or of modified methods. Both laboratories, OSL and the operator analyze the selected samples by the same analytical method such as IDMS or KED and measure also the density. Without undermining the independency of the IAEA regarding sample data, the on-going comparison intends to discover systematic errors and deficiencies of new equipment. The ITVs are applied as upper limit for OSL-OPD differences. For example, the differences in density measurement results since the regular inter-comparison started in June 2008 were always better than 1.1% defined in the Working Paper for the OSL. The differences in IDMS on concentrated Pu-samples were less than half of the 1.6% limit derived from ITV's.

6. Training and external support

Training is an important part of Quality Control Assurance at the OSL. On average, each staff of the (IAEA)-OSL participates in external training once per year, either analytical method related or Safeguards-related. Recent training sessions were for example conducted at the LSS in LaHague, ITU, Thermo-Finnigan, LANL and SAL-Seibersdorf. New (IAEA)-OSL staff participates in an introductory training at SAL Seibersdorf. In parallel, experts from European, national laboratories or manufacturers such as: EURATOM (HKED expert, DA expert), the USA (LANL, Oak Ridge, Canberra) and France (CETAMA, HKED expert) visit the OSL and regularly train the staff, provide updates on new developments and, in addition, provide technical support. Thus, new technical developments are accounted for. The participation of external experts is important also for maintaining the performance of the instruments, for example OSL receives valuable support for their HKED systems from the ITU method experts. Support in terms of reference materials such as large size dried spikes comes primary from SAL-Seibersdorf.

7. Improvements during the first three years of operation

In the first year of full operation at OSL, a number of processes, procedures and methods were improved. As key elements of the OSL QS, procedures and user requirements of the SAL-Seibersdorf were implemented. The SAL experience is beneficial at this early stage of the OSL. An example of an early improvement (beginning of 2007) at OSL was the separation chemistry in terms of IDMS: It turned out that the Working Instruction required modifications to account for the specific conditions at

the OSL-RRP. Thus, a bias in the range of ca. 1% in IDMS-Pu which was observed between April and October 2006, was resolved.

All other methods were continuously improved, e.g.: QC parameters were defined and warning/action limits derived yet to be placed in QA format, correction sheets for HKED based on QC-results were introduced, new HKED QC-samples (U-glass) were developed and are currently under testing, and for TIMS the dynamic zoom system is now used for focussing the ion beam instead of moving the Faraday cups for avoiding possible mechanical problem over the years. An important step towards information management was the establishment of a Laboratory Information Management System (LIMS) for the OSL and its databases, which is currently in the 3rd stage of development. Various applications were developed in Visual Basic by (IAEA)-OSL staff during the last years, which support the daily work such as calculating IDMS results from the raw data and are transferred into the functionalities of the LIMS.

In another important aspect, the exchange of information and the communication between (IAEA)- and (NMCC)-OSL was also continuously improved. A system for the information-exchange was established, which includes regular meetings at various levels: The daily operations in the OSL are coordinated in a joint (NMCC and IAEA) morning meeting; the activities for the next week are planned in a weekly meeting. In the monthly meeting, held together with the Japan Safeguards Office (JSGO) and via videoconference with SAL, the activities and performance results of OSL are reviewed and important developments, conclusions/proposals and strategic plans are discussed. The annual meeting summarizes achievements and performances of OSL and is a platform for strategic planning with external cooperation partners.

8. Near-future plans

In order to cope with technical developments, to further improve the methods and to reduce uncertainties, various activities are foreseen, e.g.:
(i) Under current testing and validation is a modified composition of spike for IDMS, which is expected to have lower systematic uncertainty for certain sample types. This is possible by mixing milligram amounts of highly enriched U and Pu, for which the OSL has a license to handle.
(ii) The introduction of new software, such as new data evaluation algorithms for HKED, will extend this method to a wider range of sample types and make the method more time-efficient. New software for Quality Control of TIMS at OSL is being developed by NMCC-Tokai, based on their experience in using such software, and will be introduced in the first half of 2009.
(iii) The participation in external inter-comparison and round-robin analysis will be extended to include Pu-containing samples. In that respect, it is important to establish a routine way to ship (within the A_2-limit) Pu-containing solutions to the OSL.
(iv) The statistical evaluation of the data produced at OSL, in cooperation with other Units of the Department of Safeguards, will allow us to derive and update uncertainties for the analytical methods.
(v) The analytical and auxiliary methods are regularly reviewed and technical improvements are foreseen, such as optimized processes of automated separations (2 different separation robots are installed), and improved sample/solution handling by means of better tools. In that respect, the OSL acknowledges the continuous support by the workshops of NAAL- and SAL-Seibersdorf.
(vi) Intensifying the visits of experts to the OSL and vice versa.

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POSTER SESSION

“RWP MCMS” – MEASUREMENT AND CHARACTERIZATION MEASUREMENT SYSTEM OF RADWASTE PACKAGING

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Abstract:

RWP MCMS is a system of integrated facilities that ensure the fulfillment of radioactive waste (radwaste) packaging characterization procedures in compliance with requirements for the radwaste handling at Ignalina’s NPP of Lithuania until its disposal (VATESI VD-RA-01-2001). RWP MCMS is located in a special room of the Buffer Storage equipped with a system of integrated facilities to provide adequate operating conditions for the instrumentation. This system was developed for measuring of radionuclide concentrations in the nuclear-power industry wastes in packages (~1.2×0.9×0.7 m), the ion-exchanging resin in plastic containers (1x1x1m), and the noncombustible waste in 20-feet ISO semi-containers (6,1x2,4x1,3 m).

Technique advantages:.

A measuring facility is designed for operation in an indoor room where the equivalent dose rate of the background radiation does not exceed 0.3 μSv/h. It allows measuring low-level waste ($10^2 \div 10^6$ Bq per package) with uncertainty not more than 50% (p=0.95).

Technique features:

The flux density of the volumetric source in the specific measurement geometry is calculated by double integration. For large volumetric sources with strong absorption the integration is performed not for the whole volume, but only for the part whose radiation is registered by the detector.

Measuring system consist of 6 scintillator and HPGe detectors. Application our measuring technique was certificated in 2005 at the All-Russian Institute of Physical-Technical and Radiotechnical Measurements of the Russian Agency for Standards. Since the December of 2005 as a part of spectrometric equipment “Sadovnik” produced by RPE “Doza”, software complex “InSpect” has been used at the Moscow Scientific and Industrial Association “Radon” for controlling of municipal radioactive waste. The English version of “InSpect” as a part of “Sadovnik” was delivered to the VINCA Institute of Nuclear Science of Serbia and Montenegro in September 2006.

Keywords: gamma-activity; radioactive; RadWastes; spectrometer; measuring system; nuclide-vector.

Safeguards Measures during Vitrification of HLLW at the former Karlsruhe Pilot Reprocessing Plant (WAK)

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Abstract:

The German pilot reprocessing plant "Wiederaufarbeitungsanlage Karlsruhe (WAK)" was commissioned in 1971 and finally shut down at the end of 1990. In the past the HLLW vitrification process at a dismantled reprocessing plant would not have been subject of IAEA safeguards. With the Additional Protocol in force, this is not anymore true. Concerning the safeguards relevance of the process and the glass product, the IAEA has accepted that a credible diversion path does not exist for the nuclear material. The vitrification process itself has no possibilities of recovering U and Pu and, furthermore, on the sites of FZK and WAK reprocessing capabilities do not exist anymore. Keeping this in mind, a safeguards approach has been agreed in 1999.

The practical safeguards measures have been discussed in detail within the last three years. As a result, IAEA visited VEK several times for design verification purposes and some measures are agreed as sufficient for a statement of compliance with the safeguards rules. EURATOM and IAEA will manage safeguarding VEK during the 1.5 years period of facility operation without any installed equipment. As a consequence, the IAEA will be able to spend their limited resources on other safeguards challenges.

Keywords: vitrification; HLLW; safeguards;

1. History and Political Decisions

In 1956 the (now) Forschungszentrum Karlsruhe (FZK) was founded in the north of Karlsruhe, Germany. In the beginning the aim of this nuclear research centre was the development of nuclear reactors. In relation to this work also research on reprocessing and waste handling was established. In 1967 the construction of the Wiederaufarbeitungsanlage Karlsruhe (WAK) pilot reprocessing plant started. The aim of this facility was the testing of flow sheet variations and process components developed by institutes of the FZK. Furthermore, also staff training for the planned industrial scale reprocessing plant was necessary. WAK started its hot operation in 1971. During 31 campaigns 207 Mg of uranium and 1.16 Mg of plutonium originating from different German reactors were reprocessed. The average burn-up was approx. 17 GWd/Mg U, the peak value 40 GWd/Mg U. In 1989, the German utilities decided to stop the construction of the industrial scale Wackersdorf reprocessing plant. The result was the final shut down of WAK at the end of 1990. Fig. 1 shows the actual buildings on the WAK site. Furthermore, all the research and development activities concerning reprocessing were stopped at FZK. Today, only waste conditioning and intermediate storage facilities like the Institut für Nukleare Entsorgung (INE) and the Hauptabteilung Dekontaminationsbetriebe (HDB) are in operation. As a result of a political decision by the Federal Government the HDB and all nuclear facilities which are shut down will be part of WAK GmbH from 1st July 2009.

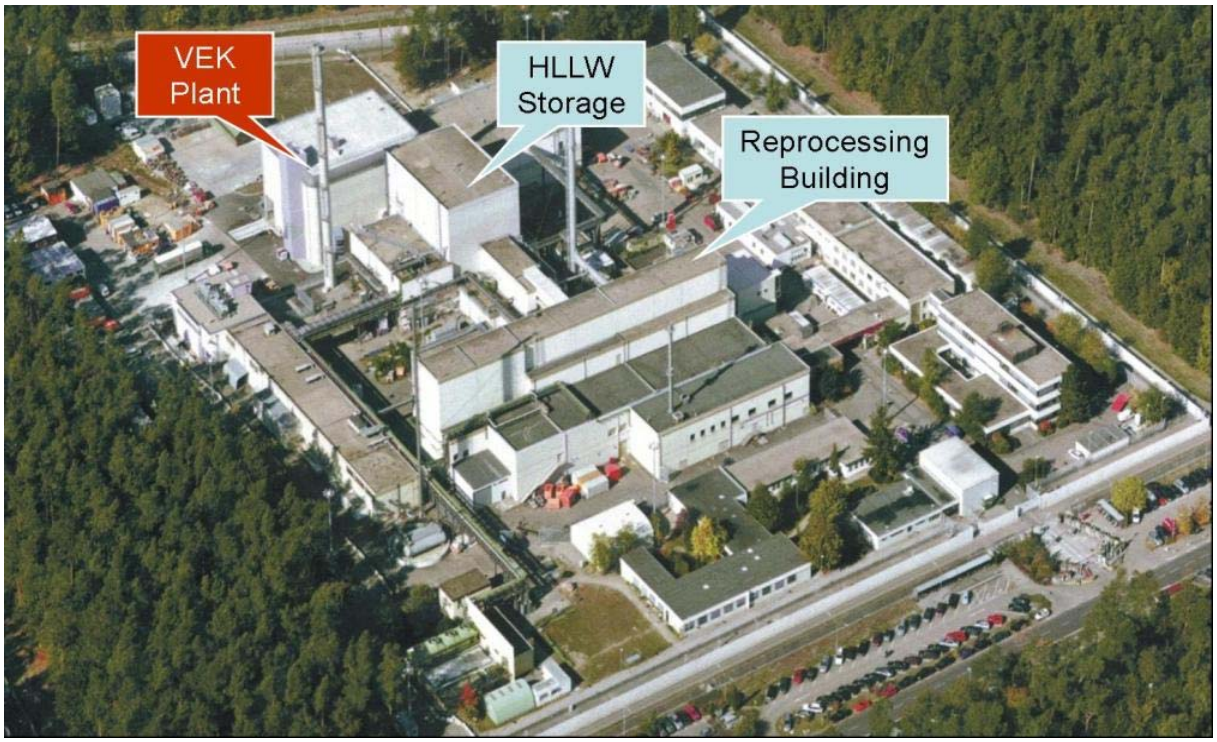

Figure 1: Aerial view on WAK site

Figure 2: Remote dismantling of the HLLW evaporator cell

Figure 3: HLLW evaporator cell after removal of the pipe penetration blocks

2. Status of the WAK Dismantling Project

After the final shut down of WAK at the end of 1990 the plant was rinsed and all separated plutonium and uranium were shipped off site. Starting in 1996, the equipment in the process building has been totally dismantled. First 12 systems only having a low activity level could be dismantled manually. In a second step the content of all process cells was dismantled by remote handling (Fig. 2) and also the necessary control systems were removed. Today even most of the pipe penetration blocks between the hot cells are already cut out (Fig. 3). In a first campaign hot spots were removed by abrasive methods. All liquid and solid wastes produced during the dismantling activities were shipped to HDB. The progress in dismantling was regularly verified during the inspections of EURATOM and the IAEA. An overview of the WAK dismantling project has been given in [1], remote dismantling is described in detail in [2].

Only one part of WAK is still in hot operation: In a separate building (LAVA) approximately 60 m³ of high level liquid waste (HLLW) with a total radioactivity of nearly 8E17 Bq are stored as “retained waste” to be conditioned on site. This vitrification project also has been established in 1996 and a new building for the Verglasungseinrichtung Karlsruhe (VEK) is finished and equipped. The cold test of the vitrification process was successful and the licence for hot operation has been granted in February 2009.

3. Design of VEK

3.1. Structure of the building

The vitrification of the HLLW is an essential step for the total dismantling and demolition of WAK. The VEK facility is only planned, constructed and licensed for this specific task that should be terminated within 1.5 years of operation. The equipment of the main process is installed in several hot cells as indicated in Fig. 4, which shows a longitudinal cross section of the VEK building. The HLLW receipt cell (1) contains two receipt tanks as well as the secondary liquid waste treatment. In the melter cell (2) the HLLW feeding vessel, the melter and the first two off-gas components (dust scrubber and condenser) can be found. The two off-gas treatment cells (wet/dry) are located behind the rear wall of the hot cells and therefore are not visible in Fig. 4.

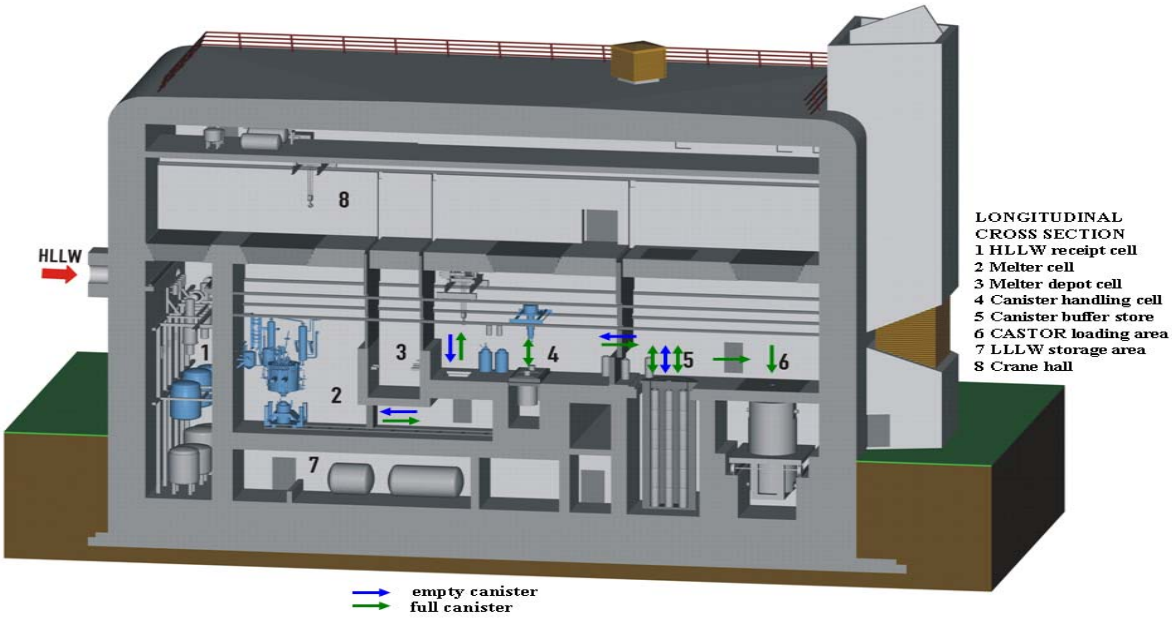

Figure 4: Longitudinal vertical cross section through the central area of the VEK building including the movement of empty and full glass canisters

In the event of a melter failure during operation, the old melter can be removed remotely and stored in the melter depot cell (3). A second melter was built and is ready for this replacement. In the canister handling cell (4) we find the cooling station, the automatic welding device for the canister lid and the decontamination unit. At the right of Fig. 4 the canister buffer store (5) and the CASTOR loading area (6) can be seen. The LLLW storage area (7) in the basement is used for liquids coming from the off-gas treatment. Heavy components in the cells may be replaced for maintenance reasons by using the equipment of the crane hall (8). The design of VEK has been already described in detail [3, 4].

3.2. Vitrification Process and Canister Handling

Looking at the flow sheet of vitrification, it is a straight forward process (Fig. 5): In LAVA a volume of 1.6 m³ HLLW is analyzed and transferred to VEK. In VEK approx. 40 l of ILLW (coming from the wet off-gas treatment) is added and the mixture is analyzed once more to verify the oxide content. The solution is transferred automatically in a small dosage vessel and than poured continuously on the surface of the melted glass. The raw glass itself is dosed batch wise as small pearls of glass frit. The HLLW is dried, calcinated and its chemical elements are incorporated as oxides into the glass matrix.

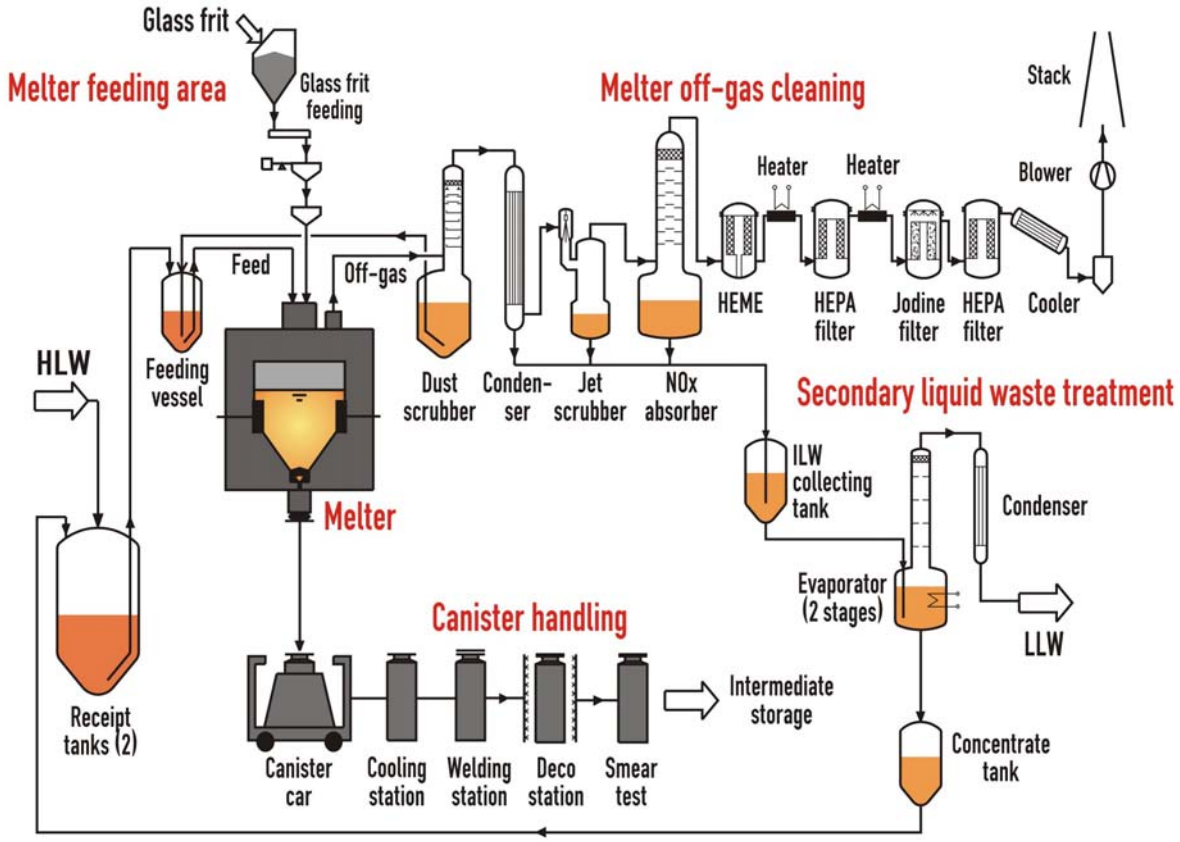

Figure 5: Simplified flow sheet of the vitrification process

The melter will contain about 400 kg of glass which will be filled every 15 hours (in four steps of 100 kg) into stainless steel canisters. The canisters are numbered and they will be weighed prior, during and after filling. The canisters will be transferred to the canister handling cell and there cooled down for several days. Then a lid is welded remotely on the canister and the canister is decontaminated. Prior to buffer storage neutron as well as beta and gamma dose rates are measured in the next cell.

The buffer storage consists of seven pipes each storing up to six canisters. The centre pipe will only be used for empty canisters so that 36 full canisters can be stored. Each 28 canisters will be loaded to a CASTOR cask. In total 130 canisters will be produced for which five CASTOR casks will be prepared and transported to an intermediate storage facility. In Fig. 4 the way of the full canisters is shown by green arrows, the way of the empty canisters by blue arrows. There are only three differences in handling: empty canisters will not arrive in a CASTOR cask, they will not be measured for activity prior to buffer storage and they will not be treated in the canister handling cell.

4. Safeguards Measures

Concerning the safeguards relevance of the process and the glass product, it is obvious that a credible diversion path does not exist for the nuclear material. The vitrification process itself has no possibilities of recovering U and Pu and, furthermore, on the sites of FZK and WAK reprocessing capabilities do not exist anymore. The nuclear material content of the HLLW is well known to the inspectorates and will be verified once more prior to vitrification. As a consequence, a safeguards approach has been agreed in 1999. It has included the design verification of VEK, sealing of a back transfer line from VEK to LAVA (which is necessary for safety reasons), the authentication of the operator measurements on the product canisters and a quarterly inspection. These measures had been recognized as sufficient for a statement of compliance with the safeguards requirements.

From the late 1990 until today two different developments can be seen:

1) Since 2005 the Regulation 302/2005 sets a focus on the fissile material control in waste. The determination of U and Pu in a large excess of fission products is a technical challenge. Although the amount of U and Pu in the HLLW of WAK is well known by the mean of multiple analyses, a single analysis may have a bias. Therefore, several checks have been introduced to secure the quality of the determination.
2) Since 2005 also the Additional Protocol is in force. At the end of 2008 the IAEA announced a positive Broader Conclusion for Germany. This enables the IAEA to reduce the verification effort for declared activities.

Keeping the IAEA informed on the VEK project by the WAK annual activity program, final discussions between the inspectorates, the German Federal Ministry of Economy and Technology, and WAK on the details of the agreed measures started in early 2006 [5]. As a result of a longer process, the following safeguards measures have been agreed:

1) Verification of the volume calibration in one of two VEK HLLW input tanks (already done in December 2006)
2) Verification of U and Pu contents by sample taking and analysis in one of two LAVA HLLW storage tanks (prior to start of hot operation in June 2009)
3) Sealing of a back transfer line from VEK to LAVA (June 2009)
4) Physical inventory taking (PIT) prior to start of hot vitrification (June 2009)
5) Quarterly inspection during hot operation of VEK with observation of the operator's determination of acidity and density on input samples
6) Information about all relevant production data of each glass canister
7) Independent neutron measurements at the CASTOR casks with the filled glass canisters
8) PIT after termination of vitrification in 2010 (possibly also an intermediate PIT will be necessary if the vitrification campaign will last more than one year)

Combining these measures, recognizing the missing reprocessing capabilities at the WAK and FZK sites and considering that the Additional Protocol is in force in the European Union, we do believe that the IAEA has sufficient qualitative and quantitative information to verify the absence of clandestine reprocessing in Germany.

5. Conclusion

The vitrification of the HLLW resulting from former nuclear fuel reprocessing is an essential step for the complete dismantling and demolition of WAK. The vitrification process has no capability of recovering fissile material. Furthermore, all research and development activities related to reprocessing have been definitely stopped at the FZK site. As a result, a credible diversion path for the fissile material in the HLLW no longer exists. With the Additional Protocol in force the IAEA gets sufficient information to verify the compliance with the safeguards criteria. As a consequence, EURATOM and the IAEA would be able to spend their limited resources on other safeguards challenges.

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Krypton-85 source term for various plutonium production schemes

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Abstract:

The operator of an illicit plutonium production facility could react to atmospheric krypton-85 air sampling, adopted for NPT verification, with an optimization of the plutonium production scheme in order to have a very low krypton-85 source term. This would make a detection of the illicit facility more difficult. In this work, by the means of burn-up calculations with different fuel enrichments and irradiation times, the krypton-85 source terms for various plutonium production schemes are assessed. The calculated krypton-85 source terms are compared with results of previous studies. It was found that the usage of depleted uranium as fuel minimizes the krypton-85 source term.

Keywords: *krypton-85; plutonium production; burn-up; source term; MCNPX*

1. Introduction

Krypton-85 is produced as many other fission products during nuclear reactor operation and remains in the fuel until reprocessing starts. Therefore, environmental sampling of this rare gas isotope has been considered by Kalinowski [1] as a tool for detecting unreported plutonium production. In order to determine the detectability of plutonium production the krypton-85 source term has to be assessed. This is defined here as the activity of krypton-85 released into the atmosphere per kilogram of plutonium separated. The important issue of this work is the question on the minimum signal that an inspector can expect under the assumption that a proliferator minimizes his krypton-85 generation in order to circumvent a krypton-85 detection. A further assumption is that for nuclear weapon production a burn-up of typically around 2 MWd/kg is used. In addition, if clandestine plutonium production takes place, the source term might be used to estimate the amount of separated plutonium.

2. Methodology of burn-up calculations

This study is based on a linkage between MCNPX and MATLAB. A reactor model is set up with MCNPX to simulate the effective cross-sections and the neutron flux. These parameters are used as an input in a self-written MATLAB code. The MATLAB code solves the set of coupled differential equations for nuclide concentrations after selected burn-up steps. All results for actinide concentrations and krypton-85 are evaluated for different enrichments of uranium-235 and compared to data from previous studies.

3. Cell burn-up calculations

The time-dependent evolution of nuclide concentrations depends on the effective cross-sections, which are influenced by macroscopic reactor parameters, like material compositions and densities, as well as the fuel temperature. The idea is to solve the burn up equations with average values for these macroscopic parameters. A characteristic magnitude for fuel consumption is defined by the burn up B [MWd/kg], which is equal to the released energy per mass of used heavy metal (HM). Finally, the

nuclide concentrations of fuel material are evaluated as a function of the burn-up according to Emendörfer and Höcker [2].

4. Depletion equations

The evolution of nuclear material composition, which is exposed to a neutron flux in nuclear power plants, is described according to Pistner [3] by time-dependent differential equations.

$$\frac{d}{dt}N^i(\vec{r},t) = -\left[\int_0^\infty dE\sigma_a^i(E)\phi(E,\vec{r},t) + \lambda^i\right]N^i(\vec{r},t) + \sum_{i\neq j}\left[f_{j\to i}(t)\int_0^\infty dE\sigma_a^j(E)\phi(E,\vec{r},t) + l_{j\to i}\lambda^j\right]N^j(\vec{r},t) \quad (1)$$

These equations, also called burn-up equations, contain the nuclide densities N^i [atoms/cm3], microscopic absorption cross-sections σ_a^i (E), σ_a^j (E) and λ as the decay constant. The formation of nuclide i due to neutron absorption of nuclide j depends on the transition probability $f_{j\to i}$ (t). The $l_{j\to i}$ is the probability for a formation of nuclide j into nuclide i by radioactive decay. As one can see two negative and two positive terms dominate equation (1). Negative terms describe the loss rate and the positive ones are formation rates. Although, the burn-up equations are valid for all types of nuclides, the most important nuclides for nuclear fuels are the actinides (Z ≥ 90), the fission products (typical: 35 ≤ Z ≤65) and structural reactor materials, like moderators and claddings (see Glaser [3]). If fission products are of interest, like krypton-85, the burn-up equations have to be expanded by one term, which is called the yield vector.

5. Modelled geometry of a LWR

The reactor, which is modelled for the burn-up calculation contains a homogeneous material composition in a unit cell geometry. Fuel pellets consist of UO_2 in various enrichments of uranium-235. The modelled cell is created according to Ascic [5] as a regular hexagon with a flat-to-flat distance of 30 cm and a height of 40 cm. More specific details for the chosen LWR are summarized in table 1. The enrichments of the fuel pellets are different, so that three source term calculations for krypton-85 are made.

Specific Power [W/g HM]	38.3 Pistner [3]	
Fuel pellet	Material	UO_2
	Density [g/cm³]	9.85 Kalinowski [1]
	Radius [cm]	0.5 Ascic [5]
Moderator	Material	H_2O
	Density [g/cm³]	01.01.00
	Enrichment of U-235 [%]	3.19; 0.7; 0.2

Table 1: Characteristics of the chosen LWR.

6. MCNPX and MATLAB calculation system

In figure 1 a scheme of the MCNPX and MATLAB calculation system is presented. First, cross-section libraries from MCNPX, materials and geometries are specified in MCNPX. After a MCNPX simulation effective cross-sections and the neutron flux are evaluated and inserted into the MATLAB code, which solves the burn-up equations for the actinides and other fission products. The numerical solution method is based on calculation slopes in defined time intervals, which have to be chosen short enough to avoid strong dependencies of effective cross-sections and neutron fluxes. After every time interval nuclide and fission product concentrations are evaluated and used as new input for MCNPX in a calculation slope. The calculation ends, when the simulated burn-up of the LWR reaches nearly 35

MWd/kg, which represents a fuel standing time of around three years of reactor operation. After that, a discharge of target material will be assumed.

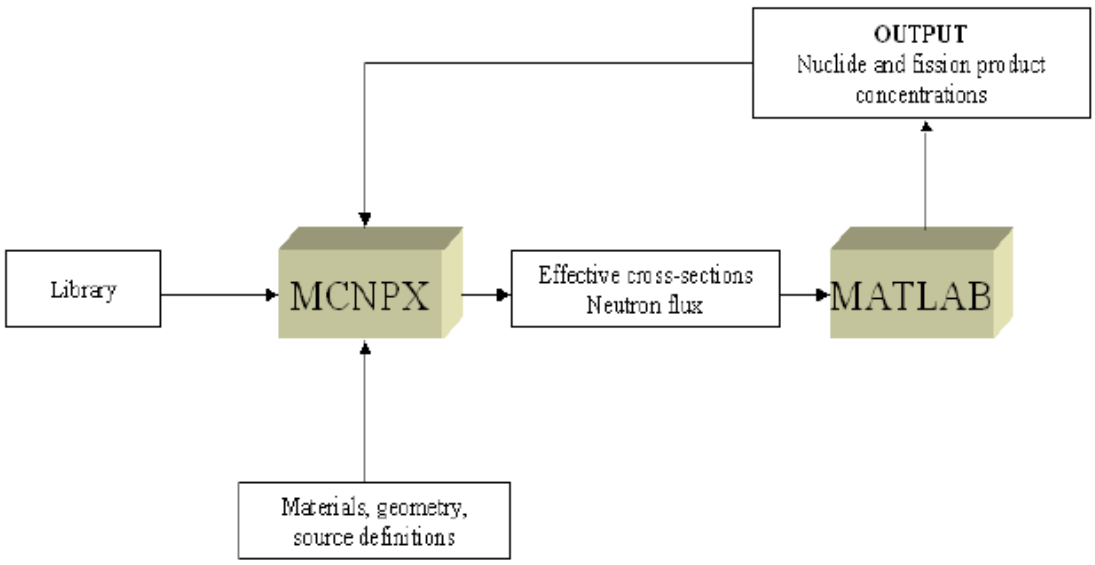

Figure 1: Scheme of the MCNPX and MATLAB calculation system.

7. Results

The calculated krypton-85 source terms [TBq/kg] without cooling decay time for the three differently enriched fuel types are summarized in the table 2. They can be compared with results of previous studies given in table 3.

Enrichment 3.19%	1.9 MWd/kg	11.1 MWd/kg	20.5 MWd/kg	29.8 MWd/kg	35.5 MWd/kg
	24.5	31.3	38.4	46.5	52.0
Enrichment 0.70%	1.7 MWd/kg	10.0 MWd/kg	20.2 MWd/kg	30.4 MWd/kg	33.9 MWd/kg
	14.7	23.4	34.1	45.2	48.9
Enrichment 0.20%	1.7 MWd/kg	10.7 MWd/kg	19.3 MWd/kg	29.7 MWd/kg	34.8 MWd/kg
	10.6	19.4	27.4	37.0	41.9

Table 2: Calculated source terms (TBq krypton-85 per kg separated plutonium) for different fuel types and burn up times.

Fuel enrichment	Low burn-up	High burn-up	Reference
LEU: 3.2 %	-	36.69 TBq/kg (33.6 MWd/kg)	Delbeke [6,7]
LEU: up to 3 %	19.45 TBq/kg	-	v. Hippel et al. [8]
NU: 0.71 %	13.59 TBq/kg	-	Delbeke [6,7]
DU: 0.2 %	-	21.10 TBq/kg (16.8 MWd/kg)	Delbeke [6,7]

Table 3: Literature values for krypton-85 source terms.

The comparison between literature and calculated source terms shows that the magnitudes are in close agreement. For high burn-up a difference of 10 TBq/kg appears for LEU (3.2%). But this difference is likely to be explained by the cooling decay time assumed by Delbeke [6,7] and by differences in the calculation systems and geometry specifications. The low-burn-up source term of 14.7 TBq/kg for natural uranium is fairly close to the value 13.6 TBq/kg determined by Delbeke [6,7].
The comparison with the source term derived[1] from data given by v. Hippel et al. [8] is not strictly possible, because no exact enrichment specification is stated. However, the assumption is reasonable that it applies for low enriched fuel with up to 3% of uranium-235. Accordingly, the krypton-85/Pu ratio of v. Hippel et al. [8] is well comparable with the result found in this study of 24.5 TBq/kg for 3.19%-LEU.

8. Conclusion on burn-up calculations

The most relevant conclusion relates to the low krypton-85 source term for depleted uranium (DU) in comparison to the other two enrichment levels. For low burn-up the specific krypton-85 activity per kilogram of plutonium produced is already lower than for high burn-up and can be further decreased by selecting depleted uranium. Therefore, this scenario is the ideal one to minimize the krypton-85 generation and circumvent a krypton-85 detection. In order to design and assess a proposed safeguards procedure, this most challenging scenario has to be used to define a performance criterion. In order to detect the production of one significant quantity of plutonium (8 kg) in a certain time, the atmospheric release of 80 TBq during this time would have to be searched for.

9. Acknowledgment

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Where will new nuclear power plants be constructed? - Validation of a predictive model

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Abstract:

Nuclear reliance (percentage of electrical generation from nuclear) has been shown to be satisfactorily predicted by a simple linear regression on various characteristics of states. It is hypothesized that nuclear deficit := actual electrical generation by nuclear minus predicted can be used to predict the degree of intent of states to rely on nuclear generation of electricity. This hypothesis is validated against two different measures of (relatively near-term) nuclear intent, in terms of an error measure developed for this nuclear deficit performs marginally satisfactorily as a predictor, but there is room for improvement. Possible sources of improvement that are suggested by a detailed analysis of the predictive errors include incorporation of a "wealth effect" in the underlying regression, and incorporation of anticipated growth in demand for electricity in the predictions from current nuclear deficits.

Keywords: electricity, predictions, power, reactors, states, statistics,

1. Introduction

The IAEA has provided [1] estimates of an answer to the titular question, as aggregated by regions, based on a survey of various member states that have indicated an interest in hosting new nuclear power plants (NPPs). This "intent-based" methodology is certainly one important way to approach the question, but official bodies and agencies have been known to misestimate capabilities. This paper comprises an alternative empirical approach. This empirical approach is based on prior work of the authors [2] that attempts to correlate current national "nuclear reliances" (fraction of electricity generated by NPPs) with various national attributes. The hope is that this "data-based" approach could lead to a more objective complement to the intent-based methodology, perhaps particularly in estimating the course of future events for states having little prior experience with civil nuclear energy. An additional characteristic of such models is their potential for exploring consequences of changes in international norms or national policies.

In more detail, this prior work is briefly summarized in the following Section 2. In Section 3 we then apply the earlier empirical model to list the number of NPPs (suitably defined) "predicted," for each of the 86 states in the data base underlying this model, along with the corresponding differences between predicted and current actual NPPs. It is hypothesized that these differences, termed as "nuclear deficits," measure the current incentive for a state to acquire additional NPPs. In Section 4 this hypothesis is applied by comparing the nuclear deficits to two presently available more-or-less objective measures of intent: NPPs under construction and NPPs planned. A novel measure of the quality of a prediction of intents, termed as "composite error," is introduced in Section 5. The composite error is then employed in Section 6 to judge quantitatively the quality of nuclear deficit as a predictor of intent, as measured by NPPs under construction and NPPs planned. Results are summarized in the concluding Section 7.

2. Summary of the empirical model

By "nuclear reliance" is intended the fraction of the electrical energy generated within a state that comes from NPPs. The empirical model stems from stepwise (ordinary least-squares) regression, with nuclear reliance as the independent variable. The underlying database consisted of the 89 nations in the world that had, in 2006, either a population of over 20 million or a GDP of over $20 billion, less three states (Afghanistan, Puerto Rico and Uganda) for which some of the requisite data were not available. The resulting 86 "nuclear candidate" (86NC) states included all except one (Armenia) that in 2007 hosted an operating NPP.

The linear model that resulted [2] from this regression ($R^2 = 0.53$, Bulgaria, France, Lithuania, Slovenia, Switzerland and Ukraine as persistent outliers) is

$$\begin{matrix} \widehat{NR} = (.30 \pm .04)IC\ ?+ (.097 \pm .032)ALGN\ ?- (.33 \pm .09)COAL - (.13 \pm .05)FCS\ ?+ (.11 \pm .04)PLTY - .034, \\ [8 \times 10^{-11}] \qquad [.004] \qquad [6 \times 10^{-4}] \qquad [.01] \qquad [.015] \end{matrix} \tag{1}$$

where $\widehat{NR}$ is predicted nuclear reliance, the estimated predictors (coefficients of the independent variables on the right) are presented as (estimate ± standard deviation), and the independent variables on the right are as follows:

- *IC?* is a (0,1)-valued measure of national effort to commercialize indigenously developed nuclear technology and finished (not raw) materials or energy on the international market. This variable was assigned a value of one for the *de jure* nuclear-weapon states, except China, plus Argentina, Belgium, Canada, Japan, Kazakhstan, Lithuania, the Netherlands, Slovakia, South Africa, South Korea and Sweden, based on a judgment that these states have significant activities in the international market for nuclear technology, including refined materials. A value of zero was assigned to the remaining 86NC states;
- *ALGN?* is a "dummy" variable, assigned value one if neither a fuel-cycle state (below) nor a *de jure* nuclear weapon state under the NPT, but either a successor state of the former Soviet Union or at one time a member of NATO, SEATO or the Warsaw Pact. Values of one also assigned to Pakistan and Taiwan. Otherwise a value of zero is assigned. This attribute is intended as an index of the historic degree of assurance that might have been perceived by a state not in a position to meet its needs for nuclear material and technology from indigenous resources;
- *COAL* is the ratio of national coal reserves (taken from [3]) to population of the state (from [4]) as normalized through division by the maximum value, over all states in the 86NC database.
- *FCS?* has value one for the fuel-cycle states, and zero otherwise. Fuel-cycle states were taken as the *de facto* nuclear-weapon states that are not nuclear weapon states (India, Israel and Pakistan), and additionally Argentina, Brazil, Canada, Japan and the Netherlands. The remaining 86NC states are assigned the value zero.
- *PLTY* is a standard political-science attribute measuring the degree to which a state has democratic tendencies as taken from a widely-used political-science data base [5], linearly renormalized to range from zero to one.

The values in brackets in (1), below an estimated predictor value, are the *p*-statistics associated to these estimates. The various independent variables are listed in the order in which they are added to the model via continued iterations of the regression process. Because selection of independent variables was based on smallest *p*-value at time of step, this tends to correspond to order of increasing associated *p*-values; however, that is not absolutely necessary, as the *p*-statistics evolve across the various steps. For example, note the inversion of *p*-values between the *ALGN?* and *COAL* variables.

3. Predicted nuclear reliances and nuclear deficits

Table 1 consists of an alphabetized list of all states in the 86NC database (first column), the number of modern NPPs predicted from the linear model (1) in the second column, the equivalent number of actual modern NPPs (third column), and the nuclear deficits (predicted minus actual) in the fourth column. Here a "modern NPP" is defined as the annual production of electrical energy from a 1000 MWe NPP, operating at 80% capacity factor. Nuclear deficits greater than one modern NPP are highlighted in green (15 states) and those less than negative one modern NPP are highlighted in red (13 states). These putatively correspond respectively to significant deficits or surpluses of capacity for civil nuclear energy, relative to mean international practice.

Table 2 contains the same data, except now the states are listed in order of decreasing nuclear deficits.

We now investigate the extent to which nuclear deficit corresponds to two-different measures of the near-term intent of the 86NC states to build new NPPs.

4. Comparison of nuclear deficits to near-term measures of intent

Here we compare estimates of nuclear intent derived from the nuclear deficits of Tables 1 and 2 against two different readily available measures of the near-term intent of states:

- *Reactors under construction*, as reported in World Nuclear Association [6]. This reference defines "under construction" as first concrete for reactor poured, or major refurbishment under way.
- *New reactors planned* +, by which we intend reactors under construction, as above, plus reactors planned, per [6]. Here "planned" means "approvals, funding or major commitment in place, mostly expected in operation within 8 years, or construction well advanced but

Table 1: Predicted and actual civil nuclear power plants (ordered alphabetically, by state).

State	NPPs (predicted)	NPPs (actual)	Nuclear deficit
Algeria	0.0	0.0	0.0
Argentina	3.4	3.0	0.4
Australia	-5.3	0.0	-5.3
Austria	0.6	0.0	0.6
Bangladesh	0.2	0.0	0.2
Belarus	0.3	0.0	0.3
Belgium	4.4	6.2	-1.9
Brazil	-3.9	2.3	-6.2
Bulgaria	1.0	2.5	-1.5
Canada	19.8	11.3	8.4
Chile	0.5	0.0	0.5
China	-8.7	5.4	-14.1
Colombia	0.2	0.0	0.2
Congo-Kinshasa	0.0	0.0	0.0
Croatia	0.3	0.0	0.3
Cuba	0.0	0.0	0.0
Czech Republic	1.5	2.2	-0.7
Denmark	0.8	0.0	0.8
Egypt	-0.2	0.0	-0.2
Ethiopia	0.0	0.0	0.0
Finland	0.7	2.5	-1.8
France	28.8	61.3	-32.5
Germany	14.0	22.8	-8.9
Ghana	0.0	0.0	0.0
Greece	1.4	0.0	1.4
Guatemala	0.1	0.0	0.1

Hong Kong	0.1	0.0	0.1
Hungary	0.8	1.8	-1.0
India	-7.4	3.4	-10.8
Indonesia	1.0	0.0	1.0
Iran	0.6	0.0	0.6
Iraq	-0.1	0.0	-0.1
Ireland	0.2	0.0	0.2
Israel	-0.3	0.0	-0.3
Italy	6.7	0.0	6.7
Japan	35.8	32.2	3.6
Kazakhstan	0.7	0.0	0.7
Kenya	0.0	0.0	0.0
Kuwait	-0.1	0.0	-0.1
Lebanon	0.0	0.0	0.0
Libya	-0.1	0.0	-0.1
Lithuania	0.9	1.5	-0.6
Malaysia	0.4	0.0	0.4
Mexico	1.9	1.3	0.7
Morocco	0.0	0.0	0.0
Myanmar	0.0	0.0	0.0
Nepal	0.0	0.0	0.0
Netherlands	3.2	0.5	2.7
New Zealand	1.0	0.0	1.0
Nigeria	0.1	0.0	0.1
North Korea	-0.1	0.0	-0.1
Norway	3.3	0.0	3.3
Pakistan	-0.5	0.3	-0.8
Peru	0.2	0.0	0.2
Philippines	0.5	0.0	0.5
Poland	2.2	0.0	2.2
Portugal	1.1	0.0	1.1
Qatar	0.0	0.0	0.0
Romania	1.3	0.8	0.5
Russia	39.1	21.2	18.0
Saudi Arabia	-0.6	0.0	-0.6
Serbia	0.3	0.0	0.3
Singapore	0.0	0.0	0.0
Slovakia	2.0	2.5	-0.5
Slovenia	0.1	0.8	-0.6
South Africa	6.2	2.0	4.3
South Korea	24.2	21.0	3.2
Spain	6.5	10.0	-3.5
Sri Lanka	0.1	0.0	0.1
Sudan	0.0	0.0	0.0
Sweden	8.2	9.0	-0.7
Switzerland	0.6	2.9	-2.3
Syria	-0.1	0.0	-0.1
Taiwan	4.9	6.6	-1.7
Tanzania	0.0	0.0	0.0
Thailand	1.2	0.0	1.2
Tunisia	0.0	0.0	0.0
Turkey	3.4	0.0	3.4
UAE	-0.1	0.0	-0.1
UK	20.0	11.2	8.8
Ukraine	2.4	11.0	-8.6
USA	181.3	115.5	65.8
Uzbekistan	0.4	0.0	0.4
Venezuela	0.7	0.0	0.7
Vietnam	-0.1	0.0	-0.1
Yemen	0.0	0.0	0.0

Table 2: Data for civil nuclear power plants (ordered by decreasing nuclear deficit)

State	NPPs (predicted)	NPPs (actual)	Nuclear deficit
USA	181.3	115.5	65.8
Russia	39.1	21.2	18.0
UK	20.0	11.2	8.8
Canada	19.8	11.3	8.4
Italy	6.7	0.0	6.7
South Africa	6.2	2.0	4.3
Japan	35.8	32.2	3.6
Turkey	3.4	0.0	3.4
Norway	3.3	0.0	3.3
South Korea	24.2	21.0	3.2
Netherlands	3.2	0.5	2.7
Poland	2.2	0.0	2.2
Greece	1.4	0.0	1.4
Thailand	1.2	0.0	1.2
Portugal	1.1	0.0	1.1
New Zealand	1.0	0.0	1.0
Indonesia	1.0	0.0	1.0
Denmark	0.8	0.0	0.8
Kazakhstan	0.7	0.0	0.7
Venezuela	0.7	0.0	0.7
Mexico	1.9	1.3	0.7
Austria	0.6	0.0	0.6
Iran	0.6	0.0	0.6
Romania	1.3	0.8	0.5
Philippines	0.5	0.0	0.5
Chile	0.5	0.0	0.5
Malaysia	0.4	0.0	0.4
Uzbekistan	0.4	0.0	0.4
Argentina	3.4	3.0	0.4
Belarus	0.3	0.0	0.3
Serbia	0.3	0.0	0.3
Croatia	0.3	0.0	0.3
Ireland	0.2	0.0	0.2
Colombia	0.2	0.0	0.2
Peru	0.2	0.0	0.2
Bangladesh	0.2	0.0	0.2
Nigeria	0.1	0.0	0.1
Hong Kong	0.1	0.0	0.1
Guatemala	0.1	0.0	0.1
Sri Lanka	0.1	0.0	0.1
Ghana	0.0	0.0	0.0
Kenya	0.0	0.0	0.0
Singapore	0.0	0.0	0.0
Congo-Kinshasa	0.0	0.0	0.0
Algeria	0.0	0.0	0.0
Ethiopia	0.0	0.0	0.0
Tanzania	0.0	0.0	0.0
Yemen	0.0	0.0	0.0
Nepal	0.0	0.0	0.0
Tunisia	0.0	0.0	0.0
Sudan	0.0	0.0	0.0
Lebanon	0.0	0.0	0.0
Myanmar	0.0	0.0	0.0
Cuba	0.0	0.0	0.0
Morocco	0.0	0.0	0.0
Qatar	0.0	0.0	0.0
Libya	-0.1	0.0	-0.1
Syria	-0.1	0.0	-0.1
Kuwait	-0.1	0.0	-0.1
North Korea	-0.1	0.0	-0.1

Iraq	-0.1	0.0	-0.1
UAE	-0.1	0.0	-0.1
Vietnam	-0.1	0.0	-0.1
Egypt	-0.2	0.0	-0.2
Israel	-0.3	0.0	-0.3
Slovakia	2.0	2.5	-0.5
Saudi Arabia	-0.6	0.0	-0.6
Slovenia	0.1	0.8	-0.6
Lithuania	0.9	1.5	-0.6
Czech Republic	1.5	2.2	-0.7
Sweden	8.2	9.0	-0.7
Pakistan	-0.5	0.3	-0.8
Hungary	0.8	1.8	-1.0
Bulgaria	1.0	2.5	-1.5
Taiwan	4.9	6.6	-1.7
Finland	0.7	2.5	-1.8
Belgium	4.4	6.2	-1.9
Switzerland	0.6	2.9	-2.3
Spain	6.5	10.0	-3.5
Australia	-5.3	0.0	-5.3
Brazil	-3.9	2.3	-6.2
Ukraine	2.4	11.0	-8.6
Germany	14.0	22.8	-8.9
India	-7.4	3.4	-10.8
China	-8.7	5.4	-14.1
France	28.8	61.3	-32.5

suspended indefinitely." Except as expressly noted otherwise, data for "reactors planned" are taken from [6].

Table 3 lists, in alphabetical order by state, the 86NC states, their respective nuclear deficits, and these two measures of near-term nuclear intent. Absence of data in the indicated sources was represented as value zero.

For some states (e.g., Algeria, Canada) the nuclear deficit seems to be an acceptable estimate of one or both of these measures of near-term nuclear intent. For other states (e.g., China, India) it appears to be a very poor predictor of either of these measures of intent. This raises the question of how to judge the quality of a predictor of any measure of nuclear intent.

We answer that question in the following Section 5, by introducing a novel error measure, termed as "composite error," that is suggested to provide such a measure of the quality of a predictor of nuclear intent, both for individual states and for ensembles in totality. This measure is then employed, in Section 6, to analysis the quality of nuclear deficit as a predictor of nuclear intent, first in aggregate over the entire ensemble consisting of the 86NC states, then for individual states. Some of the possible reasons for poor performance of nuclear deficit as a predictor for individual states are discussed in the concluding Section 7.

5. Composite error: theory

For any state (i) and measure of nuclear intent, say NI_i, we define the "composite error" of some estimate of nuclear intent, say ENI_i, as

$$CE_i = \frac{ENI_i - NI_i}{\max\left\{1.0, \left|ENI_i\right|, \left|NI_i\right|, \left|ENI_i - NI_i\right|\right\}}. \qquad (2)$$

Composite errors thus defined necessarily lie between -1 and 1, with the extremes corresponding to the nuclear deficit severely respectively underestimating or overestimating the particular measure of

Table 3: Nuclear deficits, for the 86NC states, with the two near-term measures of nuclear intent (ordered alphabetically, by state).

State	Nuclear deficit	Reactors under construction	New reactors planned +
Algeria	0.0	0	0
Argentina	0.4	0.7	1.4
Australia	-5.3	0	0
Austria	0.6	0	0
Bangladesh	0.2	0	0
Belarus	0.3	0	0
Belgium	-1.9	0	0
Brazil	-6.2	0	1.2
Bulgaria	-1.5	0	1.9
Canada	8.4	1.5	4.8
Chile	0.5	0	0
China	-14.1	8.7	33.6
Colombia	0.2	0	0
Congo-Kinshasa	0.0	0	0
Croatia	0.3	0	0
Cuba	0.0	0	0
Czech Republic	-0.7	0	0
Denmark	0.8	0	0
Egypt	-0.2	0	1
Ethiopia	0.0	0	0
Finland	-1.8	1.6	1.6
France	-32.5	1.6	1.6
Germany	-8.9	0	0
Ghana	0.0	0	0
Greece	1.4	0	0
Guatemala	0.1	0	0
Hong Kong	0.1	0	0
Hungary	-1.0	0	0
India	-10.8	3	12.8
Indonesia	1.0	0	2
Iran	0.6	0.9	2.8
Iraq	-0.1	0	0
Ireland	0.2	0	0
Israel	-0.3	0	0
Italy	6.7	0	0
Japan	3.6	2.3	17.3
Kazakhstan	0.7	0	0.6
Kenya	0.0	0	0
Kuwait	-0.1	0	0
Lebanon	0.0	0	0
Libya	-0.1	0	0
Lithuania	-0.6	0	0
Malaysia	0.4	0	0
Mexico	0.7	0	0
Morocco	0.0	0	0
Myanmar	0.0	0	0
Nepal	0.0	0	0
Netherlands	2.7	0	0
New Zealand	1.0	0	0
Nigeria	0.1	0	0
North Korea	-0.1	0	1
Norway	3.3	0	0
Pakistan	-0.8	0.3	0.9
Peru	0.2	0	0
Philippines	0.5	0	0
Poland	2.2	0	0
Portugal	1.1	0	0
Qatar	0.0	0	0

Romania	0.5	0	1.3
Russia	18.0	6	18.9
Saudi Arabia	-0.6	0	0
Serbia	0.3	0	0
Singapore	0.0	0	0
Slovakia	-0.5	0.8	0.8
Slovenia	-0.6	0	0
South Africa	4.3	0	3.6
South Korea	3.2	3	9.4
Spain	-3.5	0	0
Sri Lanka	0.1	0	0
Sudan	0.0	0	0
Sweden	-0.7	0	0
Switzerland	-2.3	0	0
Syria	-0.1	0	0
Taiwan	-1.7	0	0
Tanzania	0.0	0	0
Thailand	1.2	0	2
Tunisia	0.0	0	0
Turkey	3.4	0	2.4
UAE	-0.1	0	4.5
UK	8.8	0	0
Ukraine	-8.6	0	1.9
USA	65.8	0	15
Uzbekistan	0.4	0	0
Venezuela	0.7	0	0
Vietnam	-0.1	0	2
Yemen	0.0	0	0

nuclear intent. Composite errors less in magnitude than about .5 may be considered acceptable, as they represent errors that are, on (root-mean-square) average, smaller than the least restrictive of a factor of two in number of modern NPPs, or half of a modern NPP.

For an aggregate error measure we employ the composite-root-mean-square error,

$$CRMSE = \sqrt{\frac{1}{n}\sum_{i=1}^{n} CE_i^2}.$$

Similarly values of the composite-root-mean-square error less than about .5 are acceptable, and values near one are most unacceptable. Alternately we could employ the composite R^2 value

$$\text{comp } R^2 = 1 - CRMSE^2,$$

as an aggregate measure of goodness of fit (of the nuclear deficits to the subject measure of nuclear intent). From the preceding, values of ~ .75 are quite acceptable for the composite R^2, while values near zero are very undesirable.

As an estimate of the various measures of nuclear intent we employ here the adjusted nuclear deficit := max{0, nuclear deficit}. This is appropriate because the WNA data employed here do not take into account projected decommissioning of NPPs, so that negative values of the various measures of intent are impossible. A viable alternative, not pursued here, would be to adjust the measures of intent for those states that have made (or had externally imposed) commitments to phase out nuclear power (e.g., Germany, Lithuania, Spain, Sweden), and use nuclear deficit adjusted as max{nuclear deficit, -existing NPPs}.

6. Composite error: application

Table 4 displays the adjusted nuclear deficits, the individual composite errors, the aggregate composite R-squares, the aggregate composite root-mean-square errors and the composite R^2 for the adjusted nuclear deficits as a fit to the two near-term measures of nuclear intent previously shown in Table 3.

On the basis of the composite R^2 values (or the composite root-mean-square errors), the adjusted nuclear deficits seem to give an aggregate fit to the two measures of near-term intent that are close to acceptable Further, the fit to "reactors under construction" seems slightly better than that to "reactors planned+." However, even for "reactors under construction" the composite error reaches the maximum possible magnitude of unity for nearly one-fifth of the states considered (seventeen of the eighty six), which highlights that the fit is at best marginally acceptable.

In order better to understand the nature of this deficiency, it is useful to study more closely the distributions of the composite errors for adjusted nuclear deficit as an estimate of the two measures of nuclear intent. In Figures 1 and 2 below we approach that by displaying the composite errors in a histogram, with five equally spaced bins. These are labelled as follows: "severely underestimated" ($-1.0 \leq$ composite error < -0.6), "slightly underestimated" ($-0.6 \leq$ composite error < -0.2), "well estimated" ($-0.2 \leq$ composite error < 0.2), "slightly overestimated" ($0.2 \leq$ composite error < 0.6) and "severely overestimated" ($0.6 \leq$ composite error ≤ 1.0)

Figure 1 shows that the nearly acceptable value (0.73) of composite R^2 for "reactors under construction" as a measure of nuclear intent is attained via approximately half of the composite errors being well estimated. Of the reactors under construction not well estimated, substantially more are overestimated than are underestimated; i.e., more states are "nuclearly timid" than warranted by their adjusted nuclear deficits. Further, of those overestimated, approximately two-thirds are severely overestimated. That is, most of the nuclearly timid states are severely nuclearly timid.

If "reactors planned +" is used as the measure of intent, rather than reactors under construction, then in some aggregate sense there there is a greater nuclear intent. One might therefore expect something of a uniform shift to the left of the distribution shown in Figure 1, with the possibility of an improved (increased) value of R^2. The corresponding (slightly) smaller value (0.68) of R^2 in Table 4 shows this expectation is not met. The mechanism underlying this is perhaps best exposed through a bin-by-bin comparison of the histogram in Figure 2 to that of Figure 1. In Figure 2 slightly fewer states seem have severely overestimated nuclear intent, and about the same are slightly overestimated, which collectively in itself should result in a slightly improved composite error. However, slightly fewer have a well-estimated nuclear intent, about the same have slightly underestimated nuclear intent, and substantially more are significantly overestimated. On balance, there seems to be a net transfer from the substantially overestimated and well-estimated bins to the substantially underestimnated bin. That is, for "reactors planned +" as a measure of nuclear intent, significantly more states display a nuclearly aggressive stance than would be warranted by their current nuclear deficit.

For purposes of developing approaches (e.g.,additional independent variables) that further improve the predictive model, it potentially is useful to pinpoint the states for which the composite errors are largest in magnitude. Table 5 lists, in order of decreasing composite error, the individual states (from the 86NC database) for which the nuclear deficit severely underestimates (composite error ≤ -0.7, to one digit of accuracy) or severely overestimates (composite error ≥ 0.6, to a single digit) "reactors planned +."

Table 5 shows that the nuclearly timid states (composite error ≥ 0.6) consist almost entirely, the sole exception being the US, of states having no reactors planned or under construction. But four of these states (Italy, Mexico, UK and Poland) have proposed new NPPs, as the US has proposed more than the 15 shown as "planned+." Nonetheless, these proposed plants do not meet the rather stringent criterion used here for a plant to be considered "planned." The remaining eight (of 13) nuclearly timid states consist of states that fit the pattern of relatively developed states having little or no existing nuclear power generation, a rather small nuclear deficit (order of one or two), and no known serious proposals to construct nuclear plants. This observation suggests the possibility of some type of

Table 4: Composite errors, for nuclear deficits as a fit to the two near-term measures of nuclear intent, the 86NC states, and in aggregate.

State	Adjusted nuclear deficits	Composite error for reactors under construction	Composite error for reactors planned +
Algeria	0.0	0.0	0.0
Argentina	0.4	-0.3	-0.8
Australia	0.0	0.0	0.0
Austria	0.6	0.6	0.6
Bangladesh	0.2	0.2	0.2
Belarus	0.3	0.3	0.3
Belgium	0.0	0.0	0.0
Brazil	0.0	0.0	-1.0
Bulgaria	0.0	0.0	-1.0
Canada	8.4	0.8	0.4
Chile	0.5	0.5	0.5
China	0.0	-1.0	-1.0
Colombia	0.2	0.2	0.2
Congo-Kinshasa	0.0	0.0	0.0
Croatia	0.3	0.3	0.3
Cuba	0.0	0.0	0.0
Czech Republic	0.0	0.0	0.0
Denmark	0.8	0.8	0.8
Egypt	0.0	0.0	-1.0
Ethiopia	0.0	0.0	0.0
Finland	0.0	-1.0	-1.0
France	0.0	-1.0	-1.0
Germany	0.0	0.0	0.0
Ghana	0.0	0.0	0.0
Greece	1.4	1.0	1.0
Guatemala	0.1	0.1	0.1
Hong Kong	0.1	0.1	0.1
Hungary	0.0	0.0	0.0
India	0.0	-1.0	-1.0
Indonesia	1.0	1.0	-0.5
Iran	0.6	-0.3	-0.8
Iraq	0.0	0.0	0.0
Ireland	0.2	0.2	0.2
Israel	0.0	0.0	0.0
Italy	6.7	1.0	1.0
Japan	3.6	0.4	-0.8
Kazakhstan	0.7	0.7	0.1
Kenya	0.0	0.0	0.0
Kuwait	0.0	0.0	0.0
Lebanon	0.0	0.0	0.0
Libya	0.0	0.0	0.0
Lithuania	0.0	0.0	0.0
Malaysia	0.4	0.4	0.4
Mexico	0.7	0.7	0.7
Morocco	0.0	0.0	0.0
Myanmar	0.0	0.0	0.0
Nepal	0.0	0.0	0.0
Netherlands	2.7	1.0	1.0
New Zealand	1.0	1.0	1.0
Nigeria	0.1	0.1	0.1
North Korea	0.0	0.0	-1.0
Norway	3.3	1.0	1.0
Pakistan	0.0	-0.3	-0.9
Peru	0.2	0.2	0.2
Philippines	0.5	0.5	0.5
Poland	2.2	1.0	1.0
Portugal	1.1	1.0	1.0

Qatar	0.0	0.0	0.0
Romania	0.5	0.5	-0.6
Russia	18.0	0.7	-0.1
Saudi Arabia	0.0	0.0	0.0
Serbia	0.3	0.3	0.3
Singapore	0.0	0.0	0.0
Slovakia	0.0	-0.8	-0.8
Slovenia	0.0	0.0	0.0
South Africa	4.3	1.0	0.2
South Korea	3.2	0.1	-0.7
Spain	0.0	0.0	0.0
Sri Lanka	0.1	0.1	0.1
Sudan	0.0	0.0	0.0
Sweden	0.0	0.0	0.0
Switzerland	0.0	0.0	0.0
Syria	0.0	0.0	0.0
Taiwan	0.0	0.0	0.0
Tanzania	0.0	0.0	0.0
Thailand	1.2	1.0	-0.4
Tunisia	0.0	0.0	0.0
Turkey	3.4	1.0	0.3
UAE	0.0	0.0	-1.0
UK	8.8	1.0	1.0
Ukraine	0.0	0.0	-1.0
USA	65.8	1.0	0.8
Uzbekistan	0.4	0.4	0.4
Venezuela	0.7	0.7	0.7
Vietnam	0.0	0.0	-1.0
Yemen	0.0	0.0	0.0
CRMSE	-	0.52	0.57
comp R^2	-	0.73	0.68

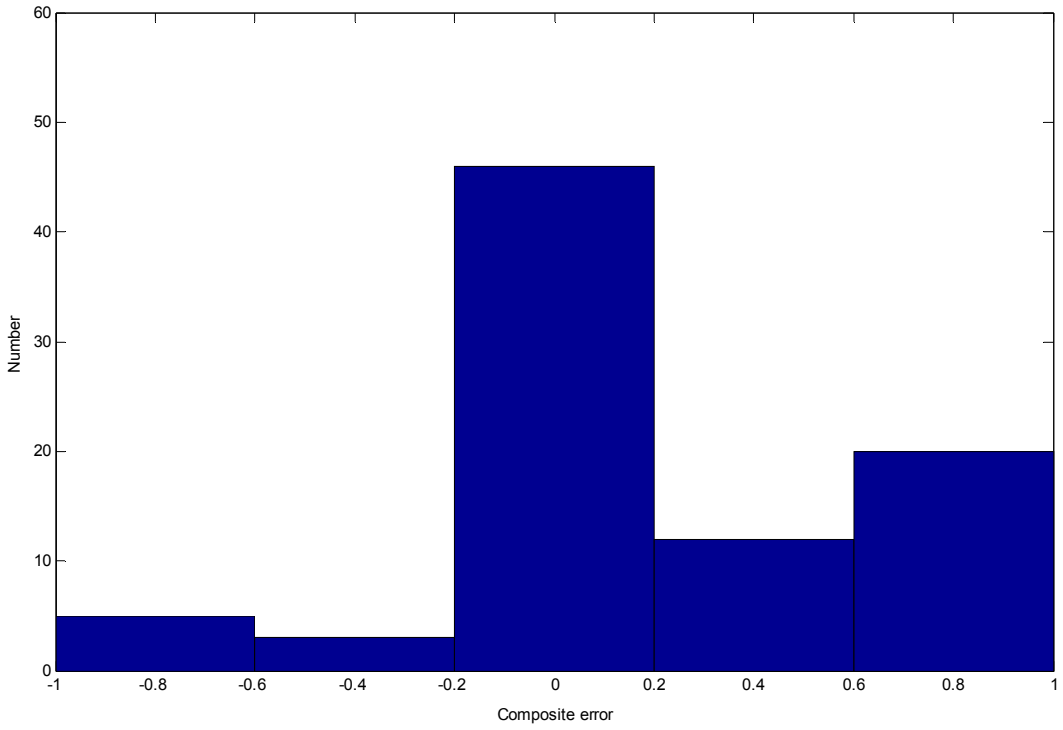

Figure 1 - Distribution of composite errors, for adjusted nuclear deficits as an estimate of reactors under construction

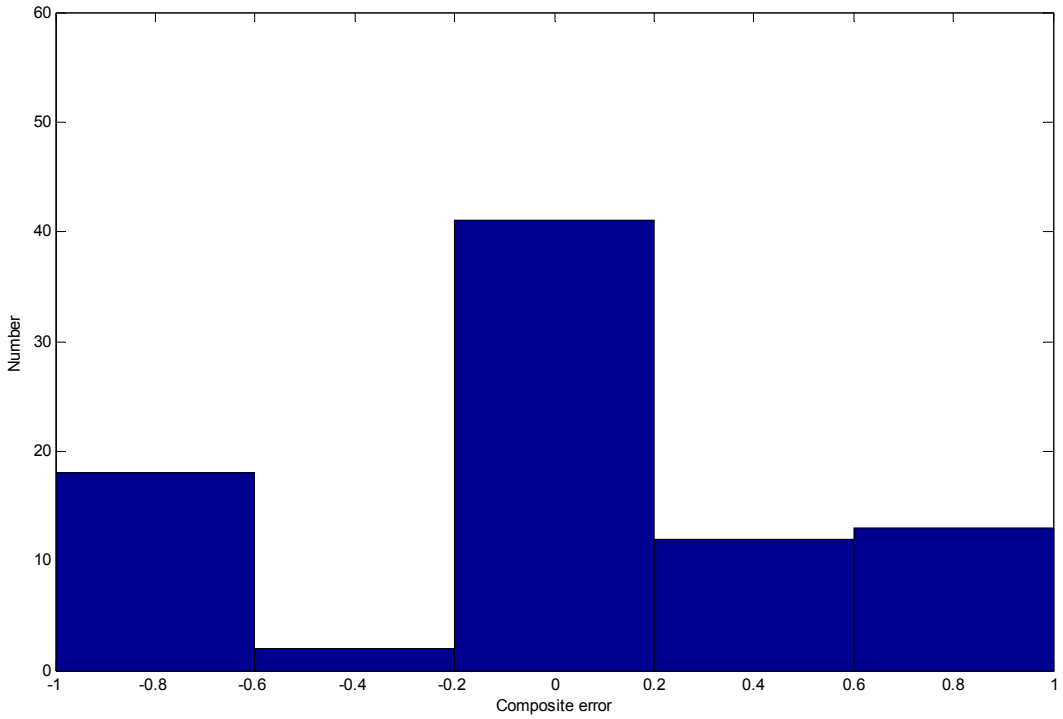

Figure 2 - Distribution of composite errors, for adjusted nuclear deficits as an estimate of reactors planned +

"wealth effect," which is to say a phenomenon under which states sufficiently wealthy choose to use some of that wealth to avoid choosing civil nuclear power.

The nuclearly aggressive states (composite error < -0.6, to one digit of accuracy) present a more complex picture. Eleven of these 17 states arguably have significant existing civil nuclear programs. A resonable conjecture is that nuclear deficit underestimates the near-term nuclear intent of these states because that intent takes into account near-term projections of their growth in electricity demand, which is not taken into account in the (current version of) the estimate provided by nuclear deficits. The remaining six nuclearly aggressive states (Egypt, Iran, North Korea, Pakistan, the UAE and Vietnam) all have relatively little experience with civil nuclear power plants. It will be interesting to see whether they succeed in finding ways to overcome this lack of experience that suffice to establish a viable civil nuclear program in the near term (next ten years).

7. Conclusion

The linear regression model developed earlier by the authors [2] has been validated as a predictor of states' intent to develop civil nuclear power. This validation was affected against two different measures of near-term nuclear intent: "reactors under construction" and "reactors planned + under construction," where the term "planned" is employed as by the World Nuclear Association [6]. A "composite error" was developed as a measure of the aggregate and individual adequacy of predictors of nuclear intent. In terms of this error measure, the "nuclear deficit" of a state (difference between nuclear reliance predicted by the regression model and actual nuclear reliance = percentage of electricity generated by nuclear plants) was shown to provide a marginally satisfactory prediction of plants under construction, and a nearly satisfactory prediction of the slightly longer term measure of intent provided by "reactors planned +."

Table 5: States for which nuclear deficit severely misestimates reactors planned +, in order of decreasing composite error relative to reactors planned +.

State	Adjusted nuclear deficits	NPPs (actual)	Reactors planned +	Composite error for reactors planned +
Greece	1.4	0.0	0	1.0
Italy	6.7	0.0	0	1.0
Netherlands	2.7	0.5	0	1.0
Norway	3.3	0.0	0	1.0
Poland	2.2	0.0	0	1.0
Portugal	1.1	0.0	0	1.0
UK	8.8	11.2	0	1.0
New Zealand	1.0	0.0	0	1.0
Denmark	0.8	0.0	0	0.8
USA	65.8	115.5	15	0.8
Venezuela	0.7	0.0	0	0.7
Mexico	0.7	1.3	0	0.7
Austria	0.6	0.0	0.6	0.6
South Korea	3.2	21.0	9.4	-0.7
Argentina	0.4	3.0	1.4	-0.8
Japan	3.6	32.2	17.3	-0.8
Slovakia	0.0	2.5	0.8	-0.8
Iran	0.6	0.0	2.8	-0.8
Pakistan	0.0	0.3	0.9	-0.9
Brazil	0.0	2.3	1.2	-1.0
Bulgaria	0.0	2.5	1.9	-1.0
China	0.0	5.4	33.6	-1.0
Egypt	0.0	0.0	1	-1.0
Finland	0.0	2.5	1.6	-1.0
France	0.0	61.3	1.6	-1.0
India	0.0	3.4	12.8	-1.0
North Korea	0.0	0.0	1	-1.0
UAE (consortium)	0.0	0.0	4.5	-1.0
Ukraine	0.0	11.2	1.9	-1.0
Vietnam	0.0	0.0	2	-1.0

Toward further improvement of predictive capability, especially for even longer term measures of nuclear intent, the following properties of the composite errors are noted. As validated against reactors under construction, the predominant source of error stems from the fact that nuclear deficit significantly overestimates, in a number of states (20 of 86; cf. Figure 1), the number of reactors presently under construction. A detailed analysis of the states for which this significant overestimation occurs indicates (data not shown in text) the corresponding states fall into one of two patterns: 12 of the 20[a] have significant activity to further develop civil nuclear power at some stage, but essentially no plants presently under construction; the remaining eight states[b] predominantly are states having little or no existing nuclear power generation, a rather small nuclear deficit (order of one or two), and no known serious proposals to construct nuclear plants. The former group suggests the hypothesis that nuclear deficit would be better suited to predict some longer-term measure of nuclear intent. The latter pattern raises the hypothesis that some "wealth effect" exists in making the choice to "go nuclear," which is to say that some relatively wealthy states choose to exercise a portion of that wealth to avoid generation of nuclear power.

The first of these hypotheses is to some extent confirmed by the validation of nuclear deficit as a predictor of reactors planned + under construction. That is, now significantly few states (13) have this measure of intent significantly overestimated by nuclear deficit. Further, only five fit the pattern of

[a] These 12 are Canada, Indonesia, Italy, Kazakhstan, Mexico, Poland, Russia, South Africa, Thailand, Turkey, UK and USA.

[b] Austria, Denmark, Greece, Netherlands, New Zealand, Norway, Portugal and Venezuela

substantial nuclear activity in the longer term; however, there remain eight that follow the hypothesized "wealth effect" pattern.

On the other hand, in aggregate nuclear deficit is a slightly poorer predictor for this somewhat longer term measure, in that it severely underestimates it for 17 states (cf. Table 5). Eleven of these 17 have significant existing nuclear programs. We hypothesize that nuclear deficit would better predict the intent of these states if it incorporated some measure of their anticipated growth in the demand for electricity. The remaining six states possibly face significant challenges in realizing their plans to establish a significant civil nuclear program.

Finally, we also hypothesize that nuclear deficit would provide a better estimate of longer-term measures of nuclear intent if the underling linear model (1), for nuclear reliance, were adjusted to account for either the apparently increased willingness of the advanced nuclear states to meet their obligation under Article IV of the NPT, or the increasing concern about climate change. These matters are deferred to future work.

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Improvements to Neutron Counting System Data Acquisition and Processing

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Abstract:

The advantages of using digital neutron pulse trains with PC based acquisition software over conventional logical TTL pulses and neutron counting hardware are well understood. The flexibility afforded by software based acquisition systems, and the scalability of the neutron counting hardware make such systems suitable for any neutron counting application. VT Nuclear Services' patented neutron electronics utilises individual detector amplifiers located directly on each detector; that are connected to local "hubs" which output digital neutron events onto a fibre optic ring which are read and stored by a PC based timestamping card. Subsequently software analysis is performed on the input digital pulse train.

VT Nuclear Services' neutron electronics extends the simple timestamping concept by additionally identifying each neutron event with the detector of origin, thus providing a full pulse stream that can be analysed in real time and/or written to file for later re-analysis. This additional detector information gives several advantages.

- *It has been utilised in distributed neutron counting systems where a single fibre optic ring can carry the data for multiple detectors in several groups.*
- *It has been used in multiplicity counting systems, allowing separate detector groups to be identified as subsets of the system (e.g. permitting ring ratio efficiency corrections).*
- *By considering which detector actually detected the neutron event, a significant improvement in the multiplicity counting statistics has been demonstrated by re-evaluation of the traditional pre-delay technique.*
- *This architecture can also accept the trigger signal from a neutron generator as an input allowing the same hardware to be utilised for active neutron applications.*

However it is the inclusions of a different type of detector that permits a potentially significant benefit to the detection limit of multiplicity counting systems. It is known that the interactions of cosmic rays with high Z items can produce a significant neutron multiplicity signal. Interactions also occur with low Z materials but typically produce only small number of neutrons. Therefore in a well shielded neutron counting chamber, the main source of coincident background neutron counts will be interactions of cosmic rays with the materials within or comprising the measurement chamber. Installing plastic scintillators around the measurement chamber and connecting them into the time stamped acquisition, allows examination of time correlation between the detection of a cosmic event passing into the chamber and resulting neutron signal from any interaction within the chamber. It is thereby possible to configure the acquisition to reject any neutron signals that are correlated with the cosmic ray event, allowing significant corrections to be applied to the measured background coincidence signal without affecting any true signal from fissile material within the chamber. This technique significantly improves the limit of detection for multiplicity counting systems.

Keywords: Neutron counting; List mode; Multiplicity; Cosmic rays

1. Introduction

VT Nuclear Services have designed, developed and patented [1][2] a neutron counting technology based upon the timestamping of pulses from the neutron detectors (note, this is often referred to as list mode data acquisition). This was developed to replace conventional scaler and shift register-based electronics previously employed in our neutron counting systems for the following reasons:

i) To permit high speed processing of neutron detector pulses, enabling the high count rate throughputs (up to 1 million counts processed per second) expected in high efficiency neutron counting systems.

ii) To provide improved noise immunity and reliability for counting systems installed in industrial facilities.

iii) To allow scalability for applications with different numbers of detectors.

iv) To provide flexibility in terms of data analysis, with the processing of the neutron "pulse train" being carried out in software.

v) To reduce costs for distributed detector systems (such as our FissTrack® Plutonium Inventory Measurement System), by minimising the amount of cabling required.

vi) To provide tamper proofing and data authentication features for Safeguards applications.

To meet these requirements, VT Nuclear Services' neutron counting electronics is based upon three proprietary components; namely head amplifiers, hub units and timestamper PC cards.

Head Amplifiers: Commercially available He3 neutron detectors are equipped with VT Nuclear Services' head amplifiers coupled directly onto the detectors. This close coupling ensures minimal susceptibility to noise interference during the amplification stage, and since the internal circuitry is potted in an electrically insulating resin, the amplifiers are effectively tamper-proof. It is noted that the amplifiers are designed to be simple low-cost items with no serviceable components.

Hub Units: The amplified signals from up to 8 head amplifiers are routed to a VT Nuclear Services' hub unit. The hub provides the high and low voltages to the amplifiers and also performs all the signal processing (pulse shaping and noise discrimination) for the signals received from the amplifiers. Significantly, the hub assigns an address to each digitised pulse which identifies the hub and detector from which the pulse originated. To minimise dead time losses, the hub derandomises pulses that arrive simultaneously to ensure that they appear as separate events in the pulse train.

Timestamper Card: The digital address data is then sent via a high speed fibre optic link to a data acquisition computer where it is received by VT Nuclear Services' timestamper unit. The timestamper appends the "time of arrival" information to each address, meaning that the detector which generated the pulse as well as the time of arrival at the timestamper is recorded. The digital pulse train is analysed within software to generate multiplicity frequency histograms which are then processed by the system software to perform either total neutron counting, coincidence counting or multiplicity analysis. The pulse train can be analysed in real time and / or written to a file for later re-analysis. The software permits detectors to be analysed individually or within user configurable groups, which greatly enhances the data analysis options.

Figure 1 shows the components of VT Nuclear Services' neutron counting electronics.

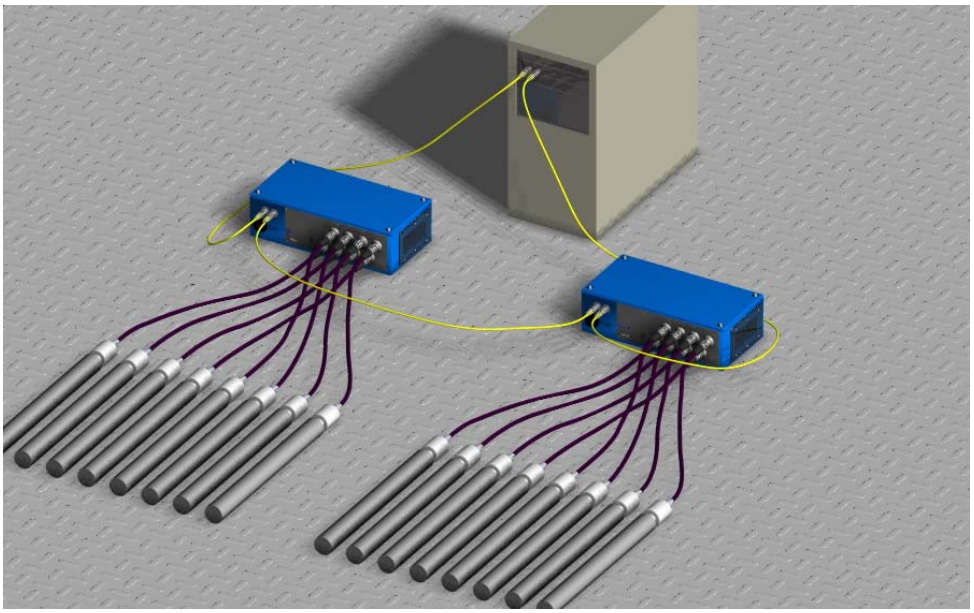

Figure 1: VT Nuclear Services' neutron counting electronics

VT Nuclear Services' neutron counting electronics extends the simple timestamping concept by additionally identifying each neutron event with the detector of origin. The recording of the detector information greatly expands the possibilities for analysis of the neutron pulse data, and several applications that have been developed by VT Nuclear Services are discussed in this report. Applications that have been applied to existing systems using the installed neutron detectors are discussed in Section 2. A recent innovative development in which a different type of detector is introduced in a Passive Neutron Coincidence Counting (PNCC) system to significantly improve the detection limit is described in Section 3.

2. Use of detector information in existing systems

2.1. Distributed detector systems

The recording of the detector of origin associated with each neutron pulse has been used to significantly reduce the cost of VT Nuclear Services' FissTrack® Plutonium Inventory Measurement System. This system is used to monitor movements of fissile material within a reprocessing plant or MOX fuel fabrication plant, and comprises a large number of neutron detectors that are distributed throughout the nuclear plant. Recording the detector of origin allows use of just a single fibre optic cable transmitting the digitised neutron pulses from many neutron detectors located in different areas of the plant. This greatly reduces the cost of installing such a system, which previously would typically require several kilometres of expensive super-screened cable.

2.2. Multiplicity / coincidence counting systems

In our TRU-D® Drum passive neutron counter (described in detail in Section 3), the additional detector information in the neutron pulse train has been used to group the neutron detectors into two sub-sets. In this system the neutron detectors are distributed as two concentric rings within the walls of the measurement chamber. The inner ring of detectors is designed to be slightly under-moderated, and hence by treating the two rings as separate groups in the analysis software, a "ring ratio" matrix correction can be determined which provides information on the moderation properties of the waste matrix.

Knowledge of where each neutron pulse originated also allows for diagnostic checking during each measurement, since the count rate in each individual detector can be compared against adjacent detectors to confirm their correct functionality. This additional data quality check supplements the pre-measurement standardisation, in which a known Cf252 neutron source is exposed to the neutron

detectors and the count rates checked, and guarantees a high level of confidence in the measurement results.

If a detector is found to be faulty, then the calibration can be adjusted offline and software re-analysis performed excluding the defective detector to generate a valid measurement result without needing to repair the fault.

2.3. Re-evaluation of the pre-delay concept

Each time a neutron event is detected in the neutron detector there is a small electronic dead time while the associated pulse is processed by the amplifier. If neutron pulses arrive at the amplifier during this period, pulse pileup can also occur due to baseline displacement of the amplifier. To reduce these dead time and pulse pileup effects, passive neutron coincidence or multiplicity counting systems incorporate a short pre-delay shift register at the input to the coincidence or multiplicity shift register circuitry. This delays the start of the “Reals plus Accidentals” coincidence counting time interval for typically 4 to 6 μs.

Although only the detector and associated amplifier in which the neutron event was detected is affected by the dead time / pulse pileup, in traditional coincidence / multiplicity neutron counting systems the pre-delay is applied to the total detector set. However, since VT Nuclear Services’ neutron counting electronics identifies the individual detector which detected the neutron, it is therefore possible to apply the pre-delay only to that detector. This is possible because the processing of the neutron pulse train and the application of the pre-delay is performed within software, and it is a simple modification to apply the pre-delay to only one detector instead of the entire detector set.

This change to the way the pre-delay is applied has been tested on the TRU-D® Drum passive neutron counter (described in Section 3), which had a totals neutron detection efficiency of approximately 25% in the configuration used for the testing. The modification to the pre-delay concept was found to increase the Reals neutron count rate measured by the system by 5%. Although this improvement may seem quite modest, it has been achieved by a simple modification to the pulse train analysis, and it would have been considerably more expensive to achieve an equivalent increase in Reals count rate via modification of the measurement chamber design.

2.4. Active neutron counting

In active neutron counting systems, the neutron counting electronics is able to accept a trigger signal from a neutron generator into the neutron pulse train, which corresponds to the firing of the neutron generator pulse. Since the origin of all the events in the pulse train is uniquely identified, the analysis software is able to extract the neutron events detected by the neutron detectors and determine when they occurred relative to the most recent neutron generator trigger signal. This therefore allows the counting electronics normally associated with active neutron counting systems to be replicated, with the neutron counts being accumulated in multi-channel or time gated scalers that are synchronised to the neutron generator pulses.

As already discussed, the recording of neutron detector address facilitates much flexibility in the data analysis. Information for different detector types such as thermal flux monitors and fast neutron detectors can easily be extracted from the pulse stream, and detector groupings can be modified within the software if necessary without recourse to hardware modification and recalibration measurements. It also allows for advanced data processing options such as imaging of the neutron signal, as implemented in VT Nuclear Services’ Imaging Passive Active Neutron (IPAN®) system. Diagnostic checks of each detector are also possible during each measurement, and should a failure of a detector be identified, then the calibration can be adjusted and re-analysis performed entirely within the software to generate a valid measurement result.

3. Use of a different detector type to improve the PNCC detection limit

The detection limit of VT Nuclear Services' TRU-D® Drum system has been significantly improved by the implementation of an innovative cosmic ray background correction methodology [3]. This is made possible by the use of our neutron counting electronics and their ability to record the detector identification as well as the pulse arrival time, which allows different types of detectors to be incorporated into a neutron counting system and contribute signals to the recorded pulse train.

The TRU-D® Drum system is a high efficiency passive neutron counting system with several advanced features to yield the best possible measurement performance. The system is designed with a close-fitting hexagonal measurement chamber and has two concentric rings of neutron detectors, giving a neutron detection efficiency of 35%. This system is also designed to have a low background through the use of background shielding and low atomic number construction materials within the measurement chamber.

Figure 2: VT Nuclear Services' TRU-D® Drum monitor

The system was developed for measurement of 200 litre drums containing Plutonium Contaminated Material (PCM) waste. However, a significant proportion of PCM waste is known to contain negligible amounts of plutonium, and could therefore be recategorised as Low Level Waste (LLW), resulting in significant cost savings with respect to subsequent waste processing and disposal. In the UK the sentencing threshold for LLW is an alpha activity content of ≤ 4 GBq/te, although a more challenging plutonium alpha activity content limit of 0.1 GBq/te is applied for disposal in the National Low Level Waste Repository near Drigg in Cumbria. These limits mean that the plutonium detection limit required to recategorise PCM waste as LLW is very low, typically being of the order of a few milligrams in a 200 litre waste drum. It is noted, however, that in order to successfully segregate LLW drums at this level, the limit of detection of the radiometric instrument needs to be considerably lower than the sentencing threshold.

The implementation of our cosmic ray background correction methodology means that segregation of LLW / PCM waste is possible with the TRU-D® Drum system using passive neutron coincidence counting (PNCC).

The detection limit of a passive neutron system is limited by the uncertainty in the measured background count rate. In order to achieve a low detection limit it is necessary to "know" the background as precisely as possible and to be certain that the "known" background is applicable for the waste item being measured. Conventionally, the measurement of a waste item is background corrected by subtracting the count rate obtained during an empty chamber measurement performed

prior to the waste item measurement. However, variations in ambient background due to external drum movements, air pressure changes and the presence of high atomic number materials in the waste drum reduce the effectiveness of such a background correction methodology.

The most significant component of the coincident (or Reals) neutron background is due to cosmic ray interactions with materials both in the waste item being measured and those used to construct the measurement chamber: this is supported by an observed strong inverse dependence of the background count rate with air pressure (see Figure 3). Note that because the system is well shielded, any neutrons produced externally to the measurement chamber are unlikely to yield genuine coincident events, and the signal shown here is due to interactions with the measurement chamber and waste item.

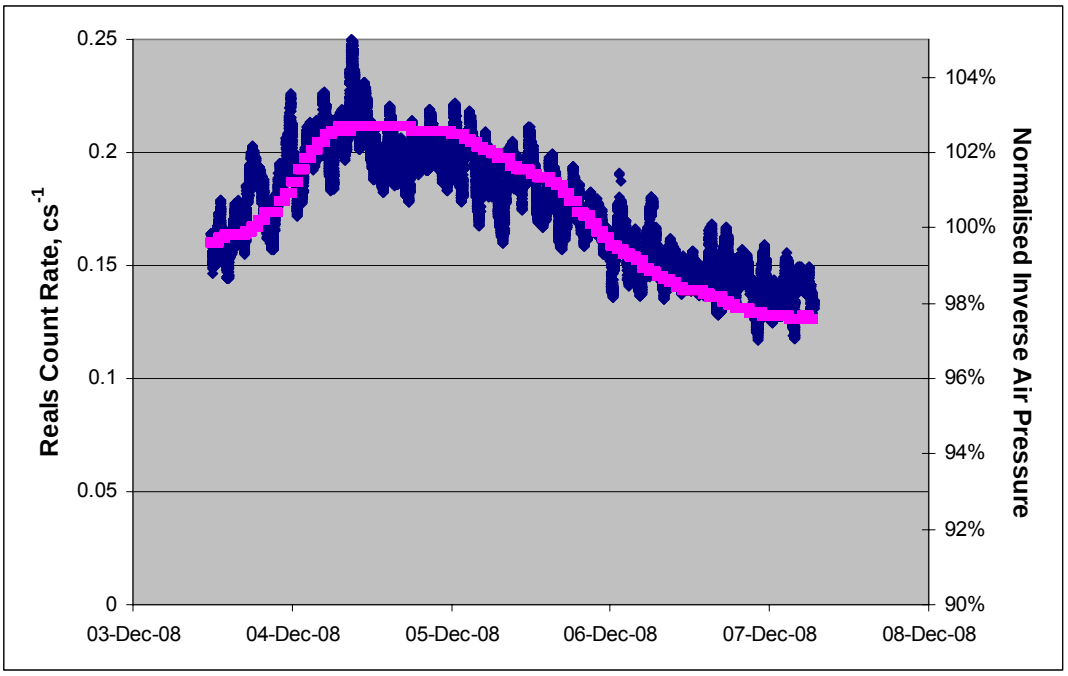

Figure 3: Comparison of background Reals neutron count rate and inverse air pressure.

The measurement data acquisition can be segmented and statistical filtering applied to the segment data to identify and remove segments that are due to noise or high multiplicity cosmic ray induced events. This technique is commonly used to reduce the effect of cosmic ray induced background; however, the large bursts of neutrons occur infrequently and this method does not account for the cosmic ray interactions that yield smaller numbers of coincident neutrons.

Commercially available plastic scintillator detectors can be used to measure the cosmic ray flux, and the measured flux also varies inversely with air pressure. The majority of the measured signal can be attributed to the muon component of the flux, and muons are known to interact with materials and produce neutrons by either spallation or capture reactions. Such reactions typically produce small bursts of neutrons which are unlikely to be filtered out by a segment rejection algorithm, but would result in an increase in the background Reals count rate. It is therefore likely that a direct measurement of the cosmic ray flux using plastic scintillators would provide useful information relating to the background during a passive neutron measurement.

By placing plastic scintillator detectors above the waste item, the cosmic ray flux incident on the item is measured. The plastic scintillators are sensitive to all of the charged particle components of cosmic rays, although the majority of the signal measured is from muons. Whilst they are also sensitive to gamma ray interactions, a suitable discriminator setting is used to remove this component of the signal. Due to the high energy of the muons, most will pass straight through the scintillator depositing a significant amount of energy, which results in a light pulse which is amplified by a photomultiplier tube to give an output that can be input directly into VT Nuclear Services' neutron counting electronics.

The particles or muons are travelling at relativistic speeds, so an interaction that occurs inside the waste item or measurement chamber can be considered to occur at the same instant it is detected in the scintillator. The resulting neutrons from both spallation and capture reactions will be promptly emitted and will therefore be detected following the normal Rossi-Alpha distribution, as shown in Figure 4.

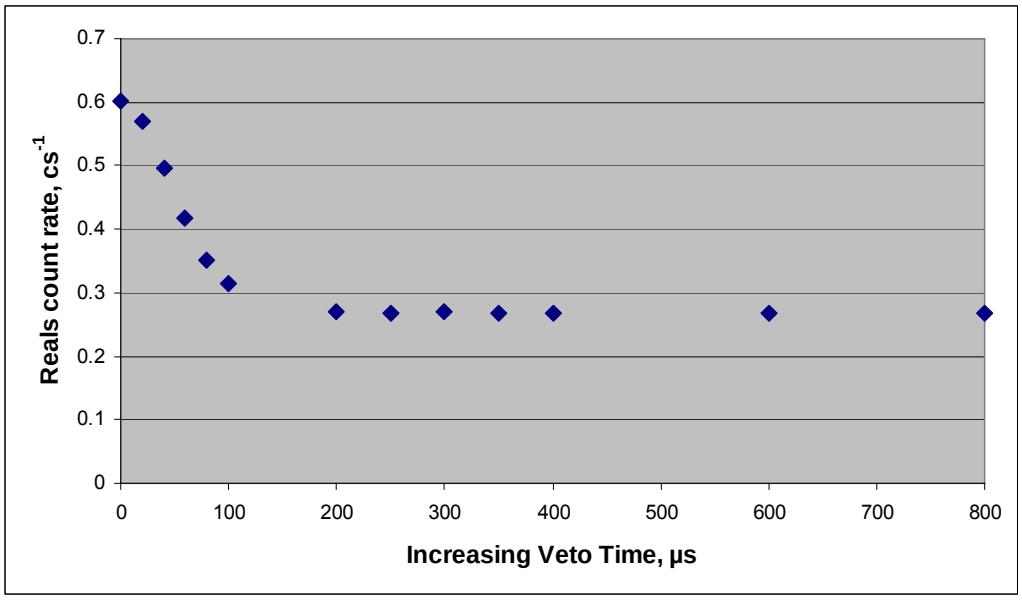

Figure 4: Effect of increasing veto time.

This has been verified by applying an anti-coincidence or "veto" approach to the detected neutron signal. Following each cosmic ray event, the neutron counter can effectively be switched off[†] and any neutrons detected within a short interval of the cosmic ray event are discarded as potentially being due to a cosmic ray interaction. The veto time is set to several times the chamber die-away time, and by vetoing in this way the measurement live time that used in subsequent calculations of the coincident neutron count rate is reduced appropriately. However, this vetoing does not affect the signal from any fissile material within the waste item, since this is not correlated to the cosmic ray flux. Hence, a significant reduction in the Reals background is achieved without biasing the Reals from fissile material, although a small increase in the measurement time will be necessary to maintain the same precision. The effect of vetoing on the neutron pulse train is illustrated in Figure 5.

[†] This is possible because VT Nuclear Services' neutron counting electronics records the detector information in the pulse train. A simple modification to the pulse train analysis software has been implemented to ignore any events recorded by the neutron detectors that are within a defined veto time of events recorded by the plastic scintillators.

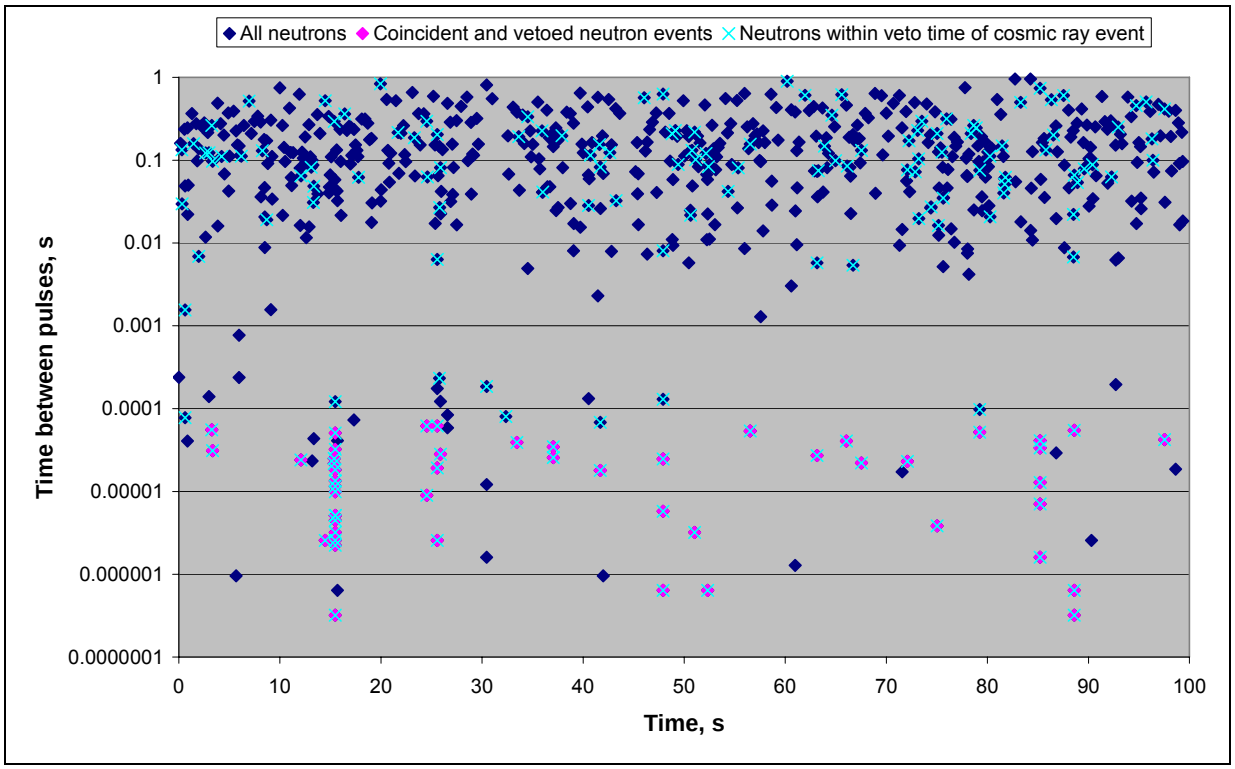

Figure 5: Cosmic ray vetoing example.

Figure 5 shows a 100 s segment of neutron count data from the recorded pulse train, with plastic scintillator detectors surrounding several lead bricks inside a neutron counting chamber. The time between successive neutron pulses is plotted against the time in the pulse train. It can be seen that the neutron pulses lie in two distinct horizontal bands: the upper band corresponds to random, single neutron events as would be expected from the Poisson distribution for a total background count rate of 4 cs^{-1}; the lower band shows neutron events detected within a short time interval of another event, and these events typically contribute to the Reals background in a PNCC measurement. The majority of these Reals background events occur as single events in the lower band (meaning that only two neutrons were detected close together), but as can be seen there are occasional vertical streaks corresponding to high multiplicity bursts of neutrons. These infrequent large burst of neutrons will normally be removed by applying a statistical filter to the time-segmented data. In the example above, approximately 2000 cosmic ray veto events per second were detected in the scintillator detectors, each resulting in a veto time of 125 µs and hence causing a system dead time of 25%. As can be seen, a resultant 25% of the random, single neutron events would not be counted due to vetoing. However, 75% of the events in the lower horizontal band have been removed by the vetoing, thus significantly reducing the Reals neutron background.

Clearly it is desirable to detect as large a fraction as possible of the incident cosmic ray flux, and to this end a detailed study was performed to determine the optimum layout of plastic scintillator detectors when mounted externally to the TRU-D® Drum measurement chamber. Initially the detectors were placed on the top of the measurement chamber, covering the whole surface. The schematic diagram in Figure 6 shows the arrangement of plastic scintillators used for the experimental trial, which allowed 25% of the chamber background Reals count rate to be correlated (i.e. vetoed) by cosmic ray detection events.

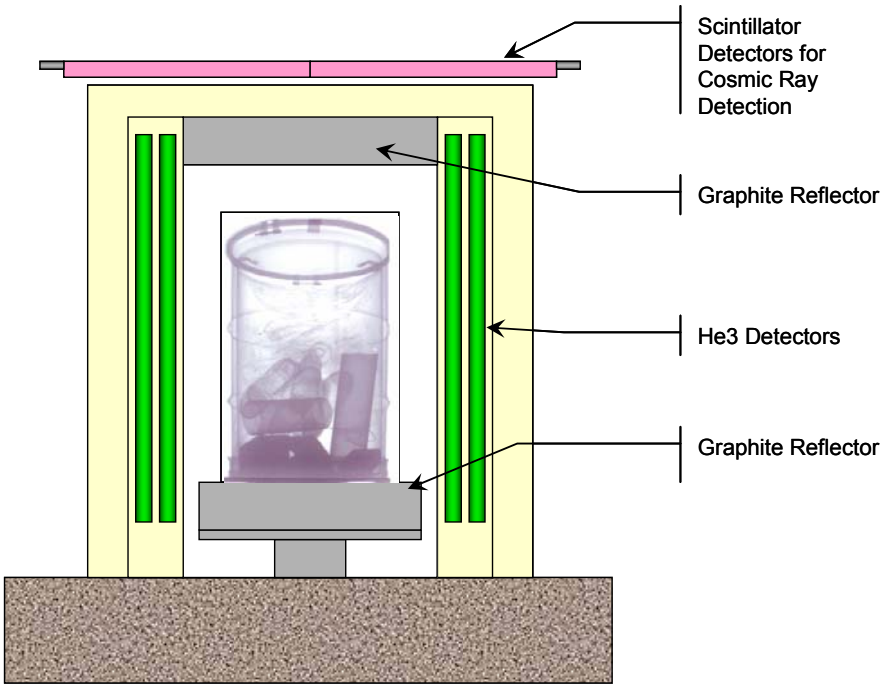

Figure 6. Schematic diagram of TRU-D® Drum showing scintillator detector arrangement used for testing

Additional experimental testing has demonstrated that positioning scintillator detectors on the sides of the measurement chamber will detect the low angle component of the cosmic ray flux and further increase the detection efficiency. It is anticipated that somewhere in the range of 65% to 90% reduction in chamber background Reals count rate could be achieved by cosmic ray veto for the TRU-D® Drum system with such an arrangement.

Figure 7 shows the improvements in detection limit that are gained in a range of different waste matrix materials using the cosmic ray background correction methodology, which vetos any neutrons detected coincidentally with cosmic ray events. Following three sets of detection limit results are included:

1. The measured results obtained with the TRU-D® Drum development system using the standard background correction method, which relies upon a separate background measurement prior to the waste item measurement.

2. The measured performance obtained with the TRU-D® Drum development system and the configuration of plastic scintillators shown in Figure 6, which gave 25% cosmic ray veto of the background Reals count rate. Note that this development PNCC system only has a neutron detection efficiency of 25% because it does not have the full complement of neutron detectors.

3. The predicted performance that could be obtained with additional plastic scintillator detectors mounted on the sides of the chamber to give a 65% cosmic ray veto of the background Reals count rate, and the full complement of He3 neutron detectors that would be present in the production system to give a neutron detection efficiency of 35%.

Note that the detection limit values shown in Figure 7 are quoted in terms of Pu240 equivalent mass (Pu240eq) at the 99% confidence level, and were obtained for the standard 20 minute passive measurement time.

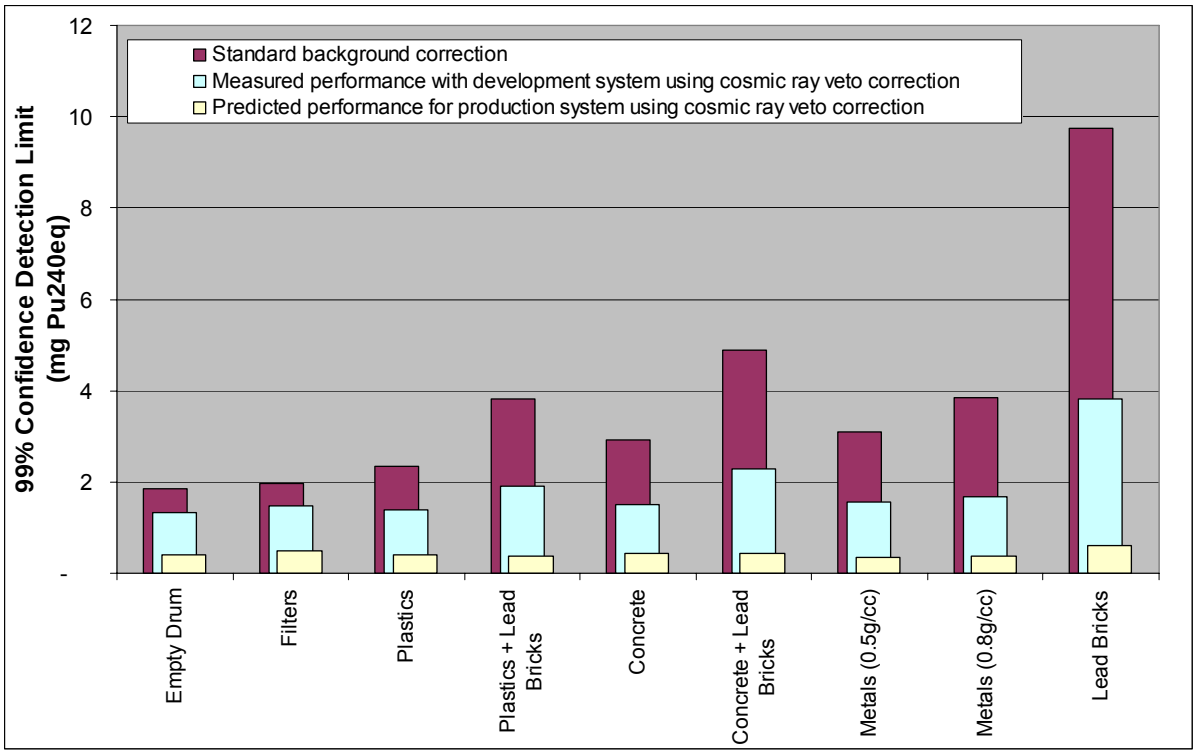

Figure 7. Improvement in Pu240eq detection limit in different waste matrices using cosmic ray vetoing.

As expected, the most significant reductions in detection limit are seen for the matrices that contain high atomic number materials such as lead. However, even in the low density matrices considerable improvement is gained.

The significant improvements in detection limit gained from using the cosmic ray vetoing background correction methodology mean that the TRU-D® Drum system is capable of segregation of LLW from PCM. This is illustrated in Table 1 where the Pu240eq mass detection limits (at 99% confidence) have been converted to a plutonium alpha activity concentration for comparison against LLW disposal limits. As it is not possible to determine the plutonium isotopic composition, a pessimistic default isotopic composition with 6% Pu240 has been used to convert the Pu240eq mass to alpha activity.

Waste Matrix (Drum Weight)	Standard background correction		Measured performance with development system using cosmic ray veto background correction		Predicted performance for production system using cosmic ray veto background correction	
	Pu240eq mass detection limit, mg	Plutonium alpha activity concentration, GBq/te	Pu240eq mass detection limit, mg	Plutonium alpha activity concentration, GBq/te	Pu240eq mass detection limit, mg	Plutonium alpha activity concentration, GBq/te
Plastics (50 kg)	2.35	2.11	1.37	1.23	0.41	0.37
Metals (120 kg)	3.09	1.16	1.55	0.58	0.36	0.13
Concrete (266 kg)	2.91	0.49	1.50	0.25	0.44	0.07
Lead (100 kg)	9.74	4.37	3.83	1.72	0.61	0.27

Table 1: TRU-D® Drum 99% confidence detection limit performance.

4. Conclusions

VT Nuclear Services have developed and patented a neutron counting electronics that generates a digital neutron pulse train containing the detector identification as well as the time each pulse was detected. This additional detector information greatly increases the possibilities for data analysis of the neutron pulse train, and also permits different information to be included such as trigger signals from a neutron generator or signals from a different detector type.

By incorporating plastic scintillator detectors into our TRU-D® Drum passive neutron counter, a pulse train is produced that includes both cosmic ray events (detected by the scintillators) and neutron events (detected by the neutron detectors). Since the Reals or coincident neutron background is dominated by cosmic ray interactions with the materials in the measurement chamber and waste item, vetoing any neutron events that closely follow cosmic ray events in the pulse stream significantly reduces this Reals background. Applying this methodology, significant reductions in the detection limit have been demonstrated using our development system which has fairly modest neutron detection efficiency (i.e. 25%) and cosmic ray veto of the background Reals count rate (i.e. 25%).

The measurements results obtained with our development system indicate that with the implementation of our cosmic ray background correction methodology, the plutonium detection limit performance of our TRU-D® Drum system will be equivalent to that of an active neutron system. This system will therefore have a sufficiently low detection limit performance to facilitate segregation of LLW from PCM.

5. References

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[2] *Improvements in and relating to monitoring of radioactive emissions*. Publication Numbers; GB2424703, EP1707992, US2006219518, JP2006284589. Priority date 31st March 2005.

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Nuclear Knowledge Management in Stakeholder Communities

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Abstract:

A possible way to increase safety and security in the use of nuclear energy nowadays is to develop shared multinational nuclear projects. There is considered multilevel stakeholder community building on international scale. We propose an interdisciplinary synergetic approach to nuclear knowledge management, based on a) self-organization (SO) of various stakeholders, b) the chaos and fuzziness concepts, and c) the principle of requisite variety. The basic hierarchical levels of stakeholder communities are proposed 1) international stakeholder community; 2) the national stakeholder community, represented by the national governments, and 3) intra-national community. Stakeholder knowledge management is considered as a key parameter of societal optimization of nuclear energy activities.

The basic elements of societal optimization are considered to be social learning, risk communication and stakeholder involvement in the decision making, thus forming a knowledge-creating community and, as the result, facilitating public acceptance of radioactive waste (RW) management policy decisions. As the basic roles of social learning in the risk communication are proposed: a) diminishing of unknown factor of perceived risk by means of familiarizing affected communities with general nuclear and specific repository issues; b) enhancing ability to understand how the community perceives all real and imaginary risks. It is proposed: self-organized social learning will promote adequate perception of risk and prevent, by diminishing uncertainties and unknown factors, social amplification of an imagined risk, as well as to increase the trust level and facilitate more adequate equity perception. As an example the education of RW management stakeholders in Latvia is described.

Keywords: stakeholder; synergetics; self-organization; knowledge; community.

1. Introduction

Nowadays, in line of a common progressively rising tendency of globalization of our world there is also developing a marked trend to globalization of nuclear energy management, aimed to increse the scale, efficiency and safety of peaceful use of nuclear energy with the final goal to contribute to further sustainable development of our Earth.

In particularly, as production fossile-based energy progressively leads to degradation of our global environment, one of relaible ways towards sustainable development seems to increase the relative weight of the nuclear energy. However, nuclear energy brings real as well as imagine risks to global security and environmental safety, controversies of various nature emerged due to the use of nuclear energy in different peaceful and military purposes.

These risks and related problems demand to develop innovative approaches with the aim to find secure, reliable and confident solutions of forthcoming use of nuclear energy and, thereby, strengthening the basic conditions of sustainable development of mankind and assuring the global society in the safety of nuclear energy.

A possible way to increase safety and security in the use of nuclear energy nowadays is to develop shared multinational nuclear projects, namely: a) construction of shared multinational spent fuel reprocessing plants, b) arrangement of advanced NPPs and research facilities, c) repositories for safe deep disposal of spent fuel and high-level RW [1].

Besides, huge complexity and specific characteristics of nuclear energy and its management modes leads to aggravated public perception of associated risks and such public perception of nuclear risks markedly differs from scientific assessment of these risks.

Such permanently growing public concern about possible nuclear risks and decision making policy

in nuclear areas – including safe disposal of the generated RW as inevitable principal back-end of any nuclear activity - may endanger forthcoming development of novel advanced projects of efficient use of nuclear energy.

Due to the global scale of advanced nuclear projects we are faced appearance of a whole set of stakeholder levels and hierarchy of their communities. Recognizing the complex problems of siting in development of any nuclear facility, as an inevitable condition for successful realization of a shared nuclear project seems to reach consensus [2] among all involved parties.

Namely, in the forthcoming problem of siting of a multinational facility, a novel essential component of stakeholder consensus building appears, namely: to reach consent – political, social, economic, ecological – among international partners, in addition to solving the whole set of intra-national consensus building items. Taking into account that international documents [3, 4] emphasize a necessity to reach shared understanding and consensus among all local communities, regions and partner countries planning to arrange shared repository for RW deep disposal [5], already nowadays we shall seek real solutions of such an interdisciplinary complex set of stakeholder consensus building,.

2. Stakeholder knowledge level – the key parameter of societal optimization of nuclear energy management

2.1 Key features and problems of the nuclear awareness

The important role of public education, distribution of all relevant information and development of communication options in the area of nuclear energy management safety has been underlined in a series binding as well as research documents. In particular, regarding the nuclear waste disposal safety, the requirement to make information on the safety of RW disposal facilities available to members of the public [3, 4, 6], by recommending the implementation of technical solutions in by taking into account public concerns [13] as well as by observing that democratization of the nuclear energy management issues includes education on all the socio-technical impacts of nuclear power [6]. Taking into account these basic statements and being completely aware of the decisive role of stakeholder learning and informing as well as society participation, we are taking hereby an attempt to analyze and describe the role of knowledge and information in the development of solutions to the complex socio-technical problems of nuclear energy activities, as the sample taking the RW management safety.

In order to develop our approach to societal optimization of nuclear energy management, let us take into account the following two characteristics, namely:
a) the public awareness and knowledge level about nuclear energy, in particular - RW management - problems is different, and
b) the inherent incompleteness in nuclear safety data, caused partly due to inherent uncertainties in the safety assessments of nuclear facilities.
On the basis of these premises, as the key elements in our approach to societal optimization we choose the approaches which could manage with these two qualities of knowledge and information.

2.2 Synergetical Approach

As nuclear energy management problem nowadays has acquired a multidisciplinary nature, thus, can be considered as an object of an interdisciplinary science – synergetics [7] being a tool for description of complex system evolution. Due to growing social activity also in the nuclear area, there is swelling such global tendency of as a shift from relations based on separation, control and manipulation towards participation, appreciation and SO [8]. It is well known that development of qualitatively novel structures is associated with SO processes - spontaneous creation of a collective order out of the local interactions between initially independent components, and basic mechanisms of SO [9] can be attributed also to information phenomena [8]. Taking into account the decisive role of information and knowledge in the management of stakeholder involvement and participation, our task could be specified as to apply the synergetics concept, namely – SO – to information and knowledge aspects [10], with the aim to consider the key societal nuclear energy management issues.

In order to propose possible routes for the arisen problem of stakeholder involvement and communication on the international scale, we will use here the recently considered synergetical approach to solve stakeholder interaction issues [18], aimed at self-organization of various stakeholder categories and their development into a harmonized stakeholder community having common strategic aims, in particular, via activating and diversifying interaction among stakeholders.

2.3 Adaptation and the principle of requisite variety

For the knowledge management aims a relevant concept to be used would be also the *principle of*

SO [11] stating that „a dynamical system, independently of its type or composition, always tends to evolve towards a state of equilibrium". Thus, our further step will based also on the adaptation concept, stating that „if consider a particular part of the original, selforganized system as the new "system", and the remainder as its "environment", then the part will be necessarily adapted to the environment" [11].

Taking such a self-organized nature of adaptation as a basic criterion of social optimization we choose the principle of requisite variety, stating: *for successful development of a given system (e.g. human being(s)) in external environment its inherent variety should exceed the variety of its environment".* In such an approach enabling us to consider the problem of social optimization as a problem of social adaptation we should to specify a real content of the meanings of: i) *external environment,* ii*) internal complexity*, iii) a *given system*.

As in our case the "given system" refers to stakeholder groups, we should try indicate possible ways how to increase the internal variety and complexity of such stakehoders in the crucially varied and complexified external environment

2. Stakeholder community, its internal complexity and external environment

Taking into account the drastic expansion and complexifying of the problem area related to nuclear energy management safety, in particular, the marked changes in the societal environment for decision making [10], let us define - in the analogy with the concept of human's three worlds [24] - the concept "external environment" as an open non-equilibrium creation. This definition includes: (1) the natural environment, (2) the social world as well as artificial environment – a set of objects, conditions and requirements emerged as the result of human and society activities. Thus, in such an extended definition the concept "external environment" will include a multitude of physical, ecological, economical, socio-cultural, psychological and other factors. Therefore, a necessary condition for successful adaptation (of the decision-making process) to a crucially changing "extended" or *external* environment and optimization of interactions with such an environment will be the predominance of humans' internal complexity over the environmental complexity.

The growing complexity of the external environment markedly displayed in conditions of the decision-making process in the RW disposal area, demands to develop advanced approaches to manage the societal requirements for RW disposal. In this task one should especially distinguish two basic factors - information and knowledge - via which we relate to our environment by self-organization processes [25].

If we take into account that the knowledge about the world contains specific components such as: a) knowledge about ourselves, and b) possible interactions between subjects [26], then one can propose a possible route of further development for the nowadays actualized problem of stakeholder involvement and communication [10, 11, 27, 28]. The aim of the proposed approach would be to reveal actual relationships between different stakeholder categories and their concerns and, as a result, to find out possible forms of self-organization of various stakeholder categories and develop them into a harmonized stakeholder community having common strategic aims. Such a joint stakeholder community including all involved parts participating in decision making is considered as the *given system* as opposed to the *external environment.*

4.Social learning and risk communication in stakeholder communities

4.1 Basic approach to social learning

Therefore, our task of optimization of stakeholder involvement can be formulated as the need to develop activities increasing the internal complexity of the joint stakeholder community. Viewing knowledge as a complexity factor, all available forms of stakeholder involvement, their education and mutual interactions can be classified as mechanisms of societal optimization, increasing the internal complexity.

First of all, it can be achieved via social and mutual learning, thereby activating and diversifying interaction between stakeholders. A key mode of this interaction can be seen as the recognition by operators and regulators of the need to contact other stakeholder groups - to increase their knowledge level as well as to enhance mutual understanding. As the knowledge itself is able to self-organize [26], the whole process of mutual learning and educating of stakeholders could emerge in a *knowledge creating stakeholder community* able to use novel [29] communication and knowledge management forms, for example – the internet - at all levels of decision-making.

4.2 Social learning using the Inernet

Due to giant complexity, these global web networks have certain SO properties, including self-adaptation to changes in operating environment, to self-healing, and – just the internet will facilitate SO of a social community in a self-organized social network. Thus**,** Internet as a modern communication networks can be considered as an important case of SO, thereby facilitating [9] information retrieval. In particular, for the case of geological repository development such web-based approach, being an advanced way for all stakeholders to access permanently updated data, has already been developed and applied with the aim to provide socially informed decision making.

4.3 Social learning and risk communication

Nowadays the role of social learning especially soundly appears in risk communication. First of all, let us note the significance of uncertainties management in confidence building in safety assessments; secondly - the decisive role of the unknown factors [30] in determining risk perception by the public; and thirdly, the ultimate significance of social learning where uncertainties can be deduced from the concept whereby the basic component of social learning – adaptation – by handling uncertainty [31] - can replenish deficiency in the necessary information.

Thus, as the perceived risk of a repository could be regarded as a function of the knowledge of repository issues [30], the role of social learning in solving risk perception issues can be demonstrated in the following way: namely, the unknown factor of perceived risk can be diminished via social learning where affected communities become familiar with nuclear issues. There is also another side of social learning, namely, the ability to understand how the community perceives all possible as well as imaginary risks.

As a valuable approach to reach such understanding one could propose a comprehensive program aimed to identify public and other stakeholder concerns. This could be achieved by increasing – via versatile communication and stakeholder involvement - the levels of such trust components [22] as openness, caring and competence. Such ability will enable these concerns to be incorporated into the decision-making mechanism, thereby raising the decision-making capacity of a stakeholder community and developing public acceptance

5. A possible approach to stakeholder communication on international scale

In the last time, a lot of efforts has been taken, with the aim to investigate an entire set of technical, economical, societal and environmental problems of RW disposal safety. For example, in a series of IAEA, OECD NEA and other documents [10-15], where, in particularly, have been considered the possible ways of stakeholder involvement as well as underlined the importance and the actual necessity to understand their concerns.

Actually, as a key mode of this interaction has been indicated, first of all, mutual learning of various stakeholder groups, aimed to elevate their knowledge level, as well as to enhance mutual understanding. In turn, the whole process of mutual learning and educating of stakeholders will emerge in a *knowledge creating stakeholder community* being capable to use novel communication and knowledge management forms, for example, the Internet, at all stages and levels of decision-making.

Such kind of web-based communication of all-level stakeholders on the international scale is expected to be especially important pertaining to the self-organization goals of stakeholders already on a multinational level as well as for promotion of activities aimed at reaching their mutual understanding and consensus building. In particular, just for the case of geological repository development such web-based approach, being a prospective way for all stakeholders to access permanently updated information, has already been markedly developed and applied "as a tool for information sharing among stakeholders with the aim to provide socially informed decision making" [19,20].

The global scale web-based communication possibilities will be especially useful tools for the development of international cooperation, in the framework of Global Nuclear Safety Regime [9], between national and intra-national stakeholders (i.e., governmental authorities in nuclear safety, operators, etc.), thereby developing functioning of international stakeholders, in particular, besides the intergovernmental organizations IAEA, OECD/NEA, such entities as:

a) multinational networks among regulatory authorities, namely:
   - International Nuclear Regulators Association (INRA),
   - Network of Regulators of Countries with Small Nuclear Programmes (NERS)

b) multinational networks among operators, such as World Association of Nuclear Operators (WANO);

c) stakeholders in international nuclear industry, such as:
- The World Nuclear Association (WNA),
- Suppliers of services and equipment,
- Non-governmental organizations, public, media.

Such multinational organizations and entities are foreseen to form a proper structural framework for development of Multilateral Nuclear Approaches (MNA), in addition (inter alia) foreseeing to create "multinational, and, in particular, regional, MNAs for new facilities based on joint ownership, drawing rights or co-management for front-end and back-end nuclear facilities"[9].

Besides, the development of partnerships between international and local organizations [21] as one of the key elements of democratic dialogue - via observing the whole set of various interests - as a prerequisite for reaching shared understanding of a disputable problem and finally promoting multi-stakeholder consensus building, taking into account a whole set of possible challenges [8], has been proposed, in particualar: i) different national legislation and time schedules, ii) transportation policy and negative public reaction being a potential source of probable disputes and controversies among partnering countries, and iii) the cost allocation.

6. Knowledge management in Latvian stakeholder communities

6.1 The Actual Necessity to Do This

To prevent further social amplification of the imagined risk for the Baldone RW repository; to provide acceptance of further siting projects; to promote wide participation in the decision making process; and to increase the transparency of radiation safety policy on the whole, Latvia has started activities related to stakeholder involvement [28]. In the RW management area the major stakeholders are local municipalities and non-governmental organizations. Recently Latvia has established legal requirements (via the *Law on EIA*, the *Regulations on the EIA procedure* and the Licensing Regulations) on the collection of opinions from stakeholders, thereby increasing the confidence of the state authorities that stakeholders will be involved in decisions.

In order to create a knowledge-based community, the national Regulatory Authority - Radiation Safety Centre (RDC) – has started regular education programs for teachers of physics as well as pupils in radiation safety issues, using lectures and published documents. To improve the public attitude towards the Baldone RW repository safety, the RDC has included on its website the early warning monitoring data, including also those of the Baldone site, thereby promoting stakeholder confidence and trust in the facility as the basic factors for an efficient solution to the social issues of RW disposal safety.

6.2 The study of the public attitude to the RW repository

Related to the issues of the present paper the most significant are studies of the public attitude to the Baldone RW repository, based on the Questionnaire of population (local residents). The Questionnaire method – direct interview, participation – voluntary; covers all farms in 3 km radius about the repository.

On the basis of the received answer one can deduce that the main factors contributing to negative attitude to repository enlargement are the following ones:
a) missing information on radiation level and on the impact on human health – the „fear factor",
b) an insufficient amount of information given to residents about the previous operation of repository including missing information on repository impact on the health of residents.

Thus, the questionnaire results allow to deduce conclusion that the basic tools which essentially could improve the situation and to weaken the negative attitude of public to repository enlargement could be recognized the safety upgrade, public education and compensation mechanisms.

6.3 The practical activities in development of stakeholder involvement.

The basic streamlines of information and education of main stakeholders (including local public) are following ones::

1. Municipality and public will be acquainted with the plan of construction of new vaults and will be regularly informed about the progress in the implementation of these plans. The main body ensuring regular flow of information – the Local Coordination Council of Repository (LCCR) being established by BAPA and including representatives of Baldone municipality, local residents as well as from regions nearby the road of waste transportation Salaspils RR – Baldone.

2. There will be elaborated communication plan, taking into account opinions of local residents using communication methods which were

successful in similar projects of other EU countries.

The planning of communication is based on following principles:

- local community and public stakeholders ensure easy access to information on planning activities and the progress in implementation,
- interactive communication process – to deal with problems worrying local residents,
- public receives answers on problematic questions being put during consultations.

7. Conclusion

Guided by recent international trends towards development of multinational RW repositories and based on contemporary non-linear science concepts, in the paper has been proposed a possible interdisciplinary approach towards stakeholder communication and building their consensus on the international scale. In line with the emphasized significance to increase internal variety of stakeholders via their social learning, tolerant communication and creative flexibility in the decision-making process, the whole hierarchical set of stakeholders community in their consensus building efforts might be called to develop and follow "the same creative multilateral engagement and active international cooperation" [35].

The proposed approach purposively should be extended also to solving similar societal-technical problems for arrangement of other multinational nuclear facilities (NPPs, research units) as well as national facilities supposedly having trans-boundary impact. On the basis of present analysis one can recommend to develop, in the frame of international cooperation projects, further systemic interdisciplinary studies preferably having goal-oriented status.

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Digital signal processing equipment for neutron coincidence counting

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Abstract:

IKI has developed four kinds of digital processing units for neutron coincidence counting. These consist of an FPGA based external unit connected via USB to a PC. One unit is essentially a virtual shift register giving the multiplicity distribution for both gates in real time.

The Pulse Train Recorder is a list mode system for recording time intervals between consecutive detector pulses. Measured data are transferred via USB line to the PC. The data acquisition program saves data in a binary file and at the same time displays time interval distribution on the screen. Calculating of total and coincidence count rates is performed post measuring by a fast evaluation software. There is also a program for calculating Rossi-alfa distribution and dye away time. There is a multichannel version of the Pulse Train Reader hardware with 16 parallel inputs. It eliminates impulse losses due to merging pulses of different preamplifiers.

Hardware and software for replaying pulse trains recorded with the list mode system has also been developed. Replaying a pulse train simulates a real source therefore it is called virtual source. It is useful for training purposes as a wide variety of virtual sources can be used anywhere without the need of any real radioactive sources on site.

Keywords: neutron coincidence counting, list mode, virtual source

1. Introduction

Neutron coincidence counting has been well established in recent decades. The method was developed for determining effective Pu-240 content of plutonium containing samples and it has been regularly used in safeguards measurements. It is based on the assumption that spontaneous fission rate is proportional to Pu-240 mass. For this first order coincidence rate called Doubles must be measured. This is made by conventional shift registers. In some cases simple multiplicity counting is not sufficient. Multiplicity counting involves also the next order moment called Triples in the calculation. Under certain assumptions the spontaneous fission rate and also induced fission and (α,n) reactions rate in the sample can be calculated.

List mode is relatively new way of neutron coincidence measurements made possible by rapid development of computational hardware. List mode devices give each incoming pulse a time stamp, and the sequence of time stamps is written to a mass storage device. Recorded pulse trains are usually evaluated after measurement. List mode devices have the advantage that the same pulse train can be analysed with different parameters or even with different codes. Another benefit of saving raw data, that in problematic cases all statistical information can be obtained from raw data, such as follow-up distribution, coincidence rates, Rossi-α distribution and dieaway time.

In the Institute of Isotopes (IKI) a series of digital processing equipment for neutron coincidence counting has been developed in the past years.

2. Virtual multiplicity spectrometer

Virtual instruments have their controls on the screen of a PC, operation intensive parts of the function are, however, implemented in hardware.

The multiplicity spectrometer acts like a conventional shift register. It simulates the entire shifting chain with predelay, R+A window, long delay and A window. For each incoming neutron multiplicity spectrum items corresponding to the population number of R+A and A windows are incremented. Resulting spectra are transferred through RS232 to the PC for calculating coincidence rates. Spectra and rates can be seen while data acquisition on the screen.

Coincidence rates needed for solving point model equations are calculated as

$$S = \frac{\sum_{i=0}^{n} (R+A)_i}{T_{meas}} = \frac{\sum_{i=0}^{n} (A)_i}{T_{meas}}$$

$$D = \frac{\sum_{i=1}^{n} i(R+A)_i - \sum_{i=1}^{n} i(A)_i}{T_{meas}}$$

$$T = \frac{\sum_{i=2}^{n} \frac{i(i-1)}{2}\left[(R+A)_i - (A)_i\right] - \frac{\sum_{i=1}^{n} i(A)_i}{S \cdot T_{meas}}\left[\sum_{i=1}^{n} i(R+A)_i - \sum_{i=1}^{n} i(A)_i\right]}{T_{meas}}$$

where S, D and T denotes Singles, Doubles and Triples rate respectively. R+A and A are measured multiplicity values for the two windows and T_{meas} is measuring time.

3. Pulse Train Recorder PTR-02

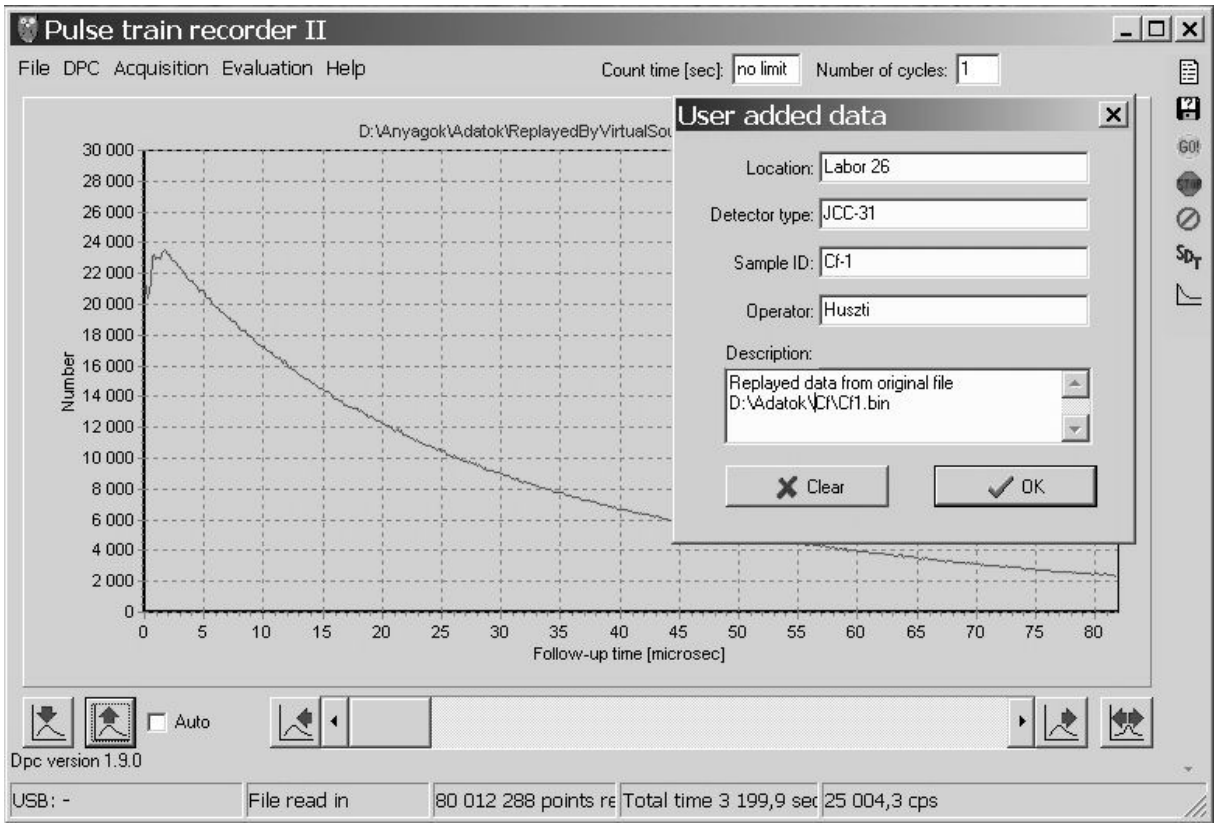
Pulse Train Recorder is a list mode device. It accepts standard TTL pulses coming from a neutron detector and measures time intervals between consecutive pulses. Collected data is sent through a USB line to a PC and stored in a binary file. Data files can be evaluated after measurement.

PTR-02 works with follow-up times instead time stamps. Time intervals between successive neutron pulses are measured as number of clock periods passed. Therefore follow-up times are integer numbers. Resulting data files contain the same information as with time stamping but require less space and can be processed faster.

Digitizing cause impulse loss if there are more incoming impulses in a clock period. Derandomizing circuits compensated for this by a fast intermediate storage which delayed excess impulse processing until the next free clock period. High clock rate of PTR-02 make this unnecessary.

Incoming neutron impulses have a statistical fluctuation but data transfer takes place in more or less uniform distributed data packets. For this reason a large buffer with appropriate control logic is needed. In order to avoid buffer overflow at high count rates, data are compressed before buffering. This multiplies effective buffer capacity up to three times at high count rates.

Data acquisition is controlled by a program running under Windows. It displays continuously follow-up distribution while data acquisition. The interactive graph is expandable or collapsible and this helps to identify possible measuring problems or interesting features in the data. Data files have a header block containing automatically filled-in and user entered information. The program can read back previously recorded binary data files and graphs.



3.1 Multichannel version

Measurements must be often made at relative high count rates. Impulse loss in the detector due to merging impulse trains from different preamplifiers is sometimes unacceptable high. In order to reduce this effect merging of impulse trains is shifted to the Pulse Train Recorder. Thank to inherent data buffering impulse trains can be merged loss free.

Multichannel hardware performed so well, that it is now used also for one channel measurements with all but one channels disabled.

3.2 Technical data

Min. follow-up time: 15 ns
Time resolution: 10 ns
Maximal count rate: 3 MHz
Output: 30 ns TTL pulse on every detected input pulse
Data output: USB 2.0 port, max. 10 MB/s
High voltage: optional

4. Evaluation software

Coincidence values and Rossi-α distribution are calculated by two separate programs, because latter requires much more time and is not always needed.

Data files consist of a header block and binary data. The header block contains three kinds of data: structure identifiers, auto filled-in fields and user entered description fields. Binary data are consecutive follow-up times in four-byte integer format.

4.1 Coincidence rate calculation

Coincidence rates are calculated by the formulas given at the virtual multiplicity spectrometer. Calculations are made in a fractional part of the acquisition time. Processing time never exceeds a few percent of acquisition time even at high count rates. The Neutron program took part in the Neutron Coincidence Benchmark Test and outraged with its speed.

Predelay, gate width and long delay can be set prior to calculation. The program can handle virtual unlimited high multiplicity numbers. The same data set can be evaluated with different parameters. Distribution values and calculation results can be saved in a text file.

4.2 Calculation of Rossi-α distribution

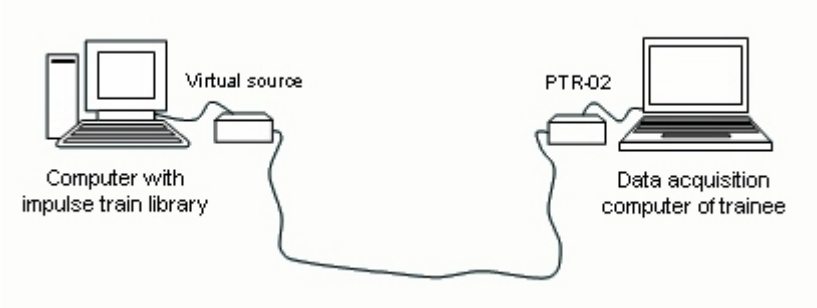
Rossi-α distribution describes the detection probability of another neutron after a trigger event in function of time. Random events have in this approach a uniform distribution whereas fission neutrons are time correlated that is usually described by a single exponential term. Calculating this distribution consumes much processor time even with integer arithmetic, that is why it is done by a separate program.

Dieaway calculation is made by fitting $N(t) = A + R \cdot e^{-t/\tau}$. The calculated distribution is 1024 μs long with 100 ns time bins. Distribution values and calculation results can be saved in a text file.

5. Virtual source

Neutron coincidence counting is one of the most important tools of nuclear safeguards. It requires good knowledge and practical experience which require not only time but different kind of neutron sources and detectors, too. There are only a few laboratories possessing all this. High efficiency detectors can be hardly moved because of their large mass and transporting radioactive sources especially nuclear ones involves a lot of administration.

With a virtual source neither a source nor a detector is needed for neutron coincidence training. A virtual source replaces source and detector by a library of previously recorded impulse trains. It reproduces previously recorded impulse trains just like they would come from a real detector. In this way impulse trains recorded in different laboratories and with different detectors can be reproduced.



The virtual source VS-02 is essentially a replayer for list mode files recorded with the Pulse Train Recorder hardware. With a generator software impulse trains with arbitrary impulse and coincidence rates can be made for replaying.

Recorded data are read back from the file, and transferred to the VS-02 unit in compressed form. The hardware stores incoming data into a FIFO buffer and processes them one by one. The decompressed value is used to set the follow-up time of the generated output pulse.

A replayed PTR-02 impulse train reproduces original follow-up times with an average error as low as 5 ns. The resulting impulse train holds all coincidence statistical properties of the original one, so not only the coincidence values S, D and T can be calculated but also Rossi-α distribution and dieaway time.

Virtual source VS-02 is based on the same FPGA platform as Pulse Train Recorder.

5.1 Technical data

Maximal rate for periodic impulses: 3 MHz
Output: 30 ns TTL impulse
Time resolution: 10 ns
Data input: USB 2.0 port, max. 10 MB/s
Power supply: 5 VDC, 2.5 A
Compatible with IAEA equipment

6. Summary

Institute of Isotopes developed on a low cost hardware platform a family of digital signal processing equipment for neutron coincidence counting.

The results show that the Pulse Train Recorder is a valuable tool for neutron multiplicity counting.

Data evaluation software for the list mode data files outraged in the ESARDA benchmark test with its speed.

The Virtual Source unit for training and education purposes is unique.

Measurement of the nuclear-material content of an inhomogeneous mixture of damaged spent fuel

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Abstract:

A passive non-destructive method was developed for determining the ^{235}U, total U and total Pu content of canisters containing an inhomogeneous mixture of spent-fuel pieces of different burn-up distributed in an irregular geometry. Using this method, the nuclear-material content of the canisters was measured, and the nuclear-material inventory of the damaged fuel was prepared and reported to EURATOM and IAEA.
The equipment constructed for the assay of the damaged fuel included high-resolution gamma-spectrometry, medium-resolution gamma spectrometry, neutron counting and gross-gamma counting devices. Here we present the results of the high-resolution gamma-spectrometric measurements.

Keywords: spent fuel; nuclear-material content; NDA

1. Introduction

In this paper we present a non-destructive method for determining the ^{235}U, total U and total Pu content of inhomogeneous mixtures of spent fuel pieces of different burn-up and irradiation history. This kind of material exists in nuclear installations worldwide, but there is no generally accepted method for verifying its nuclear-material content. Here we describe a method based on high-resolution gamma spectrometry, developed in the framework of a larger project in which the applicability of both high-resolution and medium resolution gamma-spectrometry to this problem was evaluated and complementary methods were identified for the cases where gamma-spectrometry is not suitable.

The gamma spectrometric method presented here was successfully applied for measuring the nuclear material content of the canisters containing damaged VVER-440 reactor fuel at Paks NPP in Hungary. The fuel became unusable due to damage in April 2003, and since then it has been repackaged into 72 canisters [1], [2], [3], [4], [5], which are presently kept in the spent-fuel pond, under IAEA and EURATOM safeguards. Based on these measurements, the amounts of ^{235}U, total uranium and plutonium were declared to EURATOM and IAEA for each canister individually, in compliance with the requirements of the authorities.

In addition to the nuclear measurement techniques, weight measurements have been performed for each loaded canister [6]. Therefore, an estimated value for the mass of nuclear material based on the total mass of the material loaded into the canisters was also available for each canister.

2. The damaged fuel

The spent-fuel mixture investigated in this work contained broken fuel rods, as well as pellets and parts of the cladding which fell out from the VVER-440 assemblies [7], [8], [9]. The initial inventory of nuclear material in the assemblies (i.e. before they got damaged) is known and documented. In the

fuel-damage incident, however, pellets from different types of assemblies of different burn-up and irradiation history got mixed together and now they cannot be separated nor identified.

The damaged fuel has been re-packaged into closed containers by the end of March 2007. There are three types of containers used for the remnants of the damaged assemblies [10]. Two types are used for nuclear material, and one type for the non-nuclear construction elements (i.e. assembly heads and tails). The dimensions of the containers are similar to those of the fuel assemblies, so that the loaded containers can be placed into the lattice in the spent-fuel pond. The containers with nuclear material are hereafter referred to as "canisters".

3. Measurements

3.1. Experimental setup

In this work we present results obtained with a high-resolution gamma spectrometer.

Note that simultaneously with the measurements with this spectrometer, we also did measurements by another device, registering neutron count rate, total gamma count rate and medium-resolution gamma spectra, similar to the FORK and SMOPY devices [11], [12], [13] ,[14] with several additions and enhancements. The results of the measurements with this device will be presented elsewhere.

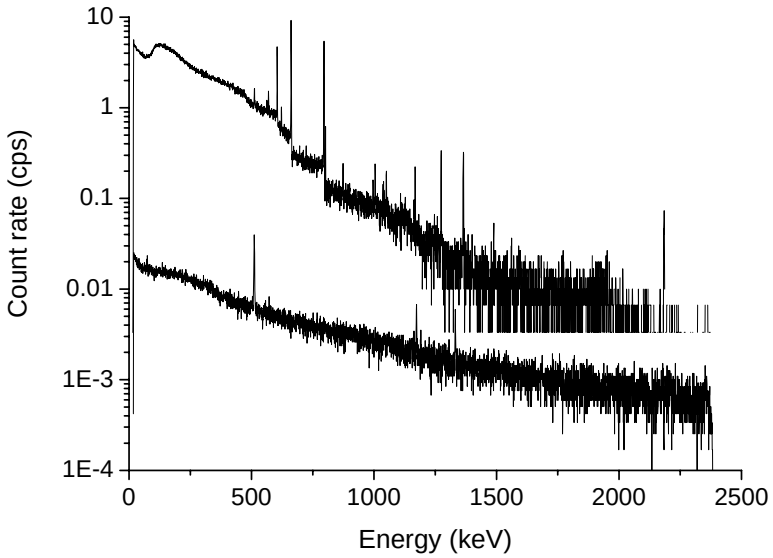


Fig. 1. High-resolution gamma spectrum of the background and of a spent fuel assembly with average burn-up of 29.766 GWd/tU

The HPGe detector was placed behind the collimator built into the concrete wall of the service pit of the reactor block. The investigated spent fuel assembly was moved up ad down under water in the service pit in front of the collimator, by the refuelling machine. The width of the collimator opening was ~20 cm, while its height was ~1 cm, making it possible to collect gamma spectrometric information with a relatively high spatial precision. The HPGe detector contained a 45 cm^3 co-axial HPGe crystal manufactured by PGT and it was connected by a ~30 m long cable to an ORTEC DSPEC Jr. 2.0 multi-channel analyzer controlled by the Gammavision software installed on a laptop computer. The laptop computer was connected through a local area network to another laptop PC, placed in the reactor hall, close to the control boot of the refuelling machine and the spectrum acquisition (e.g. acquisition start, acquisitions stop, spectrum saving) was controlled from the reactor hall over the local area network. A gamma spectrum taken with the HPGe detector is shown in Fig. 1.

For the purpose of calibration and validation in the measurement campaign for determining the nuclear-content of the canisters, in addition to the previous measurements [15] with reference assemblies in the testing phase of the project, 4 reference assemblies were also measured, several times during the measurement campaign. The assemblies were selected in such a way, that their irradiation history and cooling time was similar to that of the damaged fuel.

3.2. Scanning

The damaged-fuel canisters were scanned in both directions (up and down) from 3 sides, while the reference assemblies were scanned from 4 sides. The canisters were scanned from 3 sides because 3 is the lowest number of sides which ensures the cancellation of the geometric effects due to asymmetric positioning (see section 3.5). The assemblies were scanned from 4 sides because 4 is the lowest number of symmetric sides ensuring that both edges and faces of the assemblies are measured. Measuring the assemblies both from the edges and from the faces was important in order to see the extent of the uncertainties introduced by geometric asymmetries. The downward scanning speed was 1 mm/s and the upward speed was 1.3 mm/s. Spectrum acquisition was restarted each 100 s. In this way for each canister about 27 spectra were obtained in the downward scanning direction and 20 spectra in the upward direction on all 3 sides. There were 72 damaged-fuel canisters and 4 reference assemblies. Therefore, more than 10 thousand high-resolution gamma spectra of VVER-440 spent fuel were collected in the measurement campaign (not including the 3-4 thousand spectra taken in previous measurement campaigns in which the experimental setup was tested with regular spent fuel assemblies).

For testing purposes some of the reference assemblies and canisters were also measured at fixed height positions for 5 minutes live time. The HPGe detector was looking at the examined item through a thin collimator with a field of view of about 200x10 mm (width x height), ensuring fine spatial resolution. However, considering the inhomogeneous and unknown geometry of the damaged fuel inside the canisters, scanning was essential in order to obtain data from the entire length of the canisters.

3.3. Spectrum summing

In addition to the measured spectra taken with 100 s real time, several summed spectra were prepared. Let us denote the original measured spectra as $S_{100}(i, j, \alpha, v)$, where i=1..72 is the serial number of the canister, j=1..27 is the serial number of the scanning height, α=-120, 0 or +120 degrees is the rotational coordinate of the canister and v="up" or "down" is the scanning direction.

First of all, at each scanning height the spectra taken from 3 sides were summed up to form a spectrum with 300 s real time. This was done separately for the up- and downward scanning. In this way, a pair of spectra with 300 s real time was obtained at each scanning height - one for "up" and one for "down" – all together more than 3000 summed spectra. The count rates from these sum spectra represent an average count rate around the measured item, at a given scanning height. Let us denote these spectra as $S_{300}(i, j, v)$. That is, summing over the three rotational positions we have for each scanning height, for each scanning direction, for each canister $S_{300}(i, j, v) = \sum_{\alpha} S_{100}(i, j, \alpha, v)$

Next, for both scanning directions a "big" sum was prepared by adding up all spectra measured in the same scanning direction. In this way two summed spectra were obtained for each measured item (canister or assembly) with ~30 and ~45 minutes real time for upward and downward scanning, respectively. Let us denote the summed spectrum corresponding to the i-th canister as $S_{canister}(i, v)$. Then summing over all scanning heights we have for each scanning direction, for each canister $S_{canister}(i, v) = \sum_{j} S_{300}(i, j, v)$

Finally, all spectra from all canisters measured in a given direction were added together. This way we got 2 spectra: one containing the sum of all measurements in the downward direction, and one in the upward direction, for the total damaged spent-fuel amount. We denote these spectra as $S_{total}(v)$. Then we have the two spectra $S_{total}(down) = \sum_{i} S_{canister}(i, down)$ and $S_{total}(up) = \sum_{i} S_{canister}(i, up)$, one for "up" and one for "down".

3.4. Data processing

Spectrum summing and extracting the relevant information from the spectra was done using software specialy developed for this purpose at the Institute of Isotopes, Budapest, Hungary. From all summed and all the ~10 thousand measured high-resolution gamma spectra the following information was extracted using the software developed:

Peak area and count rate of:
- Cs-137 662 keV peak
- Cs-134 605, 796 and 1365 keV peaks
- Co-60 1332 keV peak
- Eu-154 1275 keV peak
- Ce-Pr-144 2186 keV peak

Activity ratios (evaluated using relative efficiency calibration):
- Cs-134 / Cs-137
- Eu-154 / Cs-134
- Eu-154 / Cs-137

Ratio of Cs-134 1365 to 605 keV peaks
Ratio of Cs-137 662 keV to the Cs-134/Cs-137 activity ratio

The peak areas were evaluated by summing the counts in the regions of interest. From these peak areas the above mentioned activity ratios were determined by relative efficiency calibration using the peaks of ^{134}Cs to construct the relative efficiency curve. In addition, the ^{134}Cs/^{137}Cs activity ratio was also calculated by the CsRatio code [16] [17], which uses a sophisticated peak fitting algorithm to determine the peak areas.

The software developed at the Institute of Isotopes was also used for quick drawing of various profiles of the investigated objects, e.g. 662 keV counts profile, ^{134}Cs/^{137}Cs activity ratio profile, mass profile etc. A screen shot of the software in operation can be seen in Fig. 2.

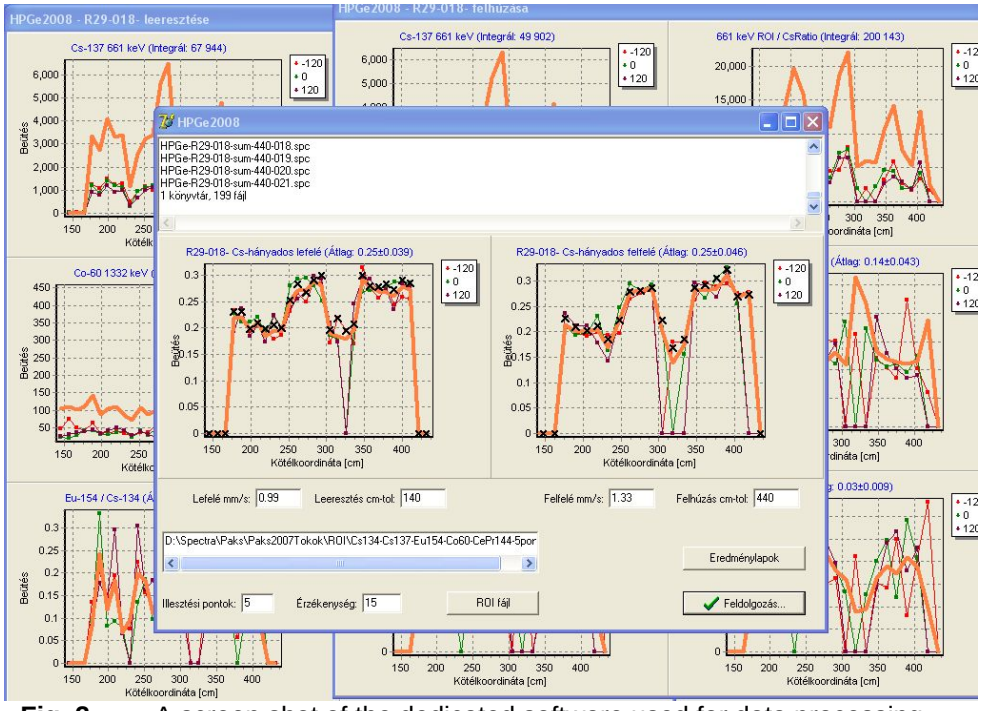
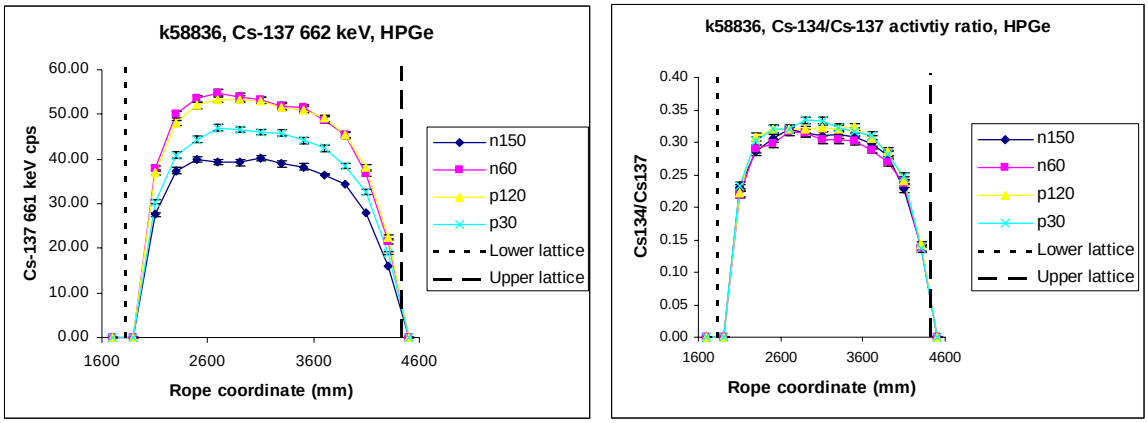


Fig. 2. A screen shot of the dedicated software used for data processing

3.5. Dealing with asymmetries

The ^{137}Cs 662 keV count rate profile and the ^{134}Cs/^{137}Cs activity ratio profile of an assembly, taken by the HPGe detector from 4 sides of the assembly are shown in Fig. 3.



a) b)

Fig. 3. HPGe profiles of an assembly with average burn-up of 24.057 GWd/tU: a) ^{137}Cs 662 keV count rate profile; b) ^{134}Cs/^{137}Cs activity ratio profile determined by relative efficiency calibration

It can be seen from the gamma profiles that the count rates of the 662 keV line are different for the four sides of the assembly and they are much lower at the edges than at the faces of the assembly. Based on very detailed measurements at different source-to-detector distances and measurements in 5 degree steps around a reference assembly, it was established that the main reasons for these differences are of geometrical nature. These geometrical effects include, e. g., bad positioning accuracy (originating from deviations of the assemblies from a vertical straight line) and increased absorption at the edges of the assemblies. The geometric effects produce 5 to 30 % differences in the 662 keV count rates for the different sides of the assembly. The eventual asymmetric burn-up of the assemblies in most cases only accounts up to ~3 % differences, as indicated by the ^{134}Cs/^{137}Cs profiles, which do not depend on the measurement geometry, since they are calculated using relative efficiency calibration.

Taking the averages of any 3 or 4 of the above mentioned detailed measurements of the 662 keV count rate at symmetric positions to each other, one obtains that the averages differ from each other merely about 1 %. This confirms that using the sum spectra representing the average count rates at each height cancels out the differences which come from geometrical asymmetries.

4. Determining the apparent burn-up of the spent-fuel mixture

It is assumed that the activity of ^{137}Cs is proportional to the burn-up of the nuclear material, while the activity of ^{134}Cs is proportional to the square of the burn-up. This implies that the count rate of the 662 keV line of ^{137}Cs and also the activity ratio ^{134}Cs/^{137}Cs should be proportional to the burn-up. We determined the activity ratio using relative efficiency calibration, therefore it is not sensitive to the changes of the measurement geometry.

In particular, the activity ratio determined from the downward scanning summed spectra of the reference assemblies was related to the average burn-up of the assemblies supplied by the operator as

$$BU = kR$$

$$R := \frac{A(Cs134)}{A(Cs137)} \exp[(\lambda_{Cs134} - \lambda_{Cs137})t] \quad , \tag{1}$$

where *BU* is the average burn-up, *A(Cs134)/A(Cs137)* is the ^{134}Cs/^{137}Cs activity ratio determined by relative efficiency calibration, *t* is the cooling time, the λ-s are the corresponding decay constants and we have obtained by measurement that for the investigated VVER-440 assemblies *k*=83.7 GWd/tU.

5. Determining the mass of nuclear material in spent fuel

In this work determining the mass of nuclear material in spent fuel is based on the principle that the mass of nuclear material, m, is proportional to the product of the concentration, ρ, of that particular type of nuclear material (^{235}U, U-total or PU-total) and of the total spent fuel mass, m_{fuel}, in the observed volume:

$$\begin{aligned} m(U_{total}) &= \rho(U_{total})m_{fuel} \\ m(U235) &= \rho(U235)m_{fuel} \\ m(Pu) &= \rho(Pu)m_{fuel} \end{aligned} \quad . \qquad (2)$$

Here under "spent-fuel mass" (m_{fuel}) we mean the sum of the masses of the remaining uranium, fission products and minor actinides. This mass is, apart from the mass defect, the same as the total mass of uranium in fresh fuel of the same volume. Therefore, equation set (2) is, in fact, equivalent to the definition of the concentration ρ, which is defined as the ratio of the mass of a particular type of nuclear material to the initial uranium mass. In this work the dimensionless quantity ρ is given as the percentage of the initial uranium mass.

The concentration of nuclear material, ρ, is calculated from known correlations between the concentration and the burn-up (see section 5.1), while the (apparent) burn-up is determined from the ^{134}Cs/^{13}7Cs activity ratio (section 4). Finally, the spent-fuel mass is determined from the count rate of the 662 keV peak of ^{137}Cs, normalized to the ^{134}Cs/^{13}7Cs activity ratio (see section 5.2 below).

5.1. Determining the concentration of nuclear material

By “concentration” of nuclear material here we mean the ratio of the mass of ^{235}U, ^{238}U and total Pu to the total mass of the spent fuel (which is practically the same as the initial mass of uranium). The concentration of nuclear material in the investigated spent fuel is determined from the correlations between the burn-up and the amounts of ^{235}U, ^{238}U and total Pu in the spent fuel. We used the correlations obtained by depletion calculation codes for VVER-440 assemblies at Paks NPP, given in reference [18]. In the burn-up range of interest for us, the concentration of the nuclear material in the spent fuel is calculated using the formulas

$$\begin{aligned} \rho(U_{total}) &= k_0(U_{total}) + k_1(U_{total})BU + k_2(U_{total})BU^2 \\ \rho(U235) &= k_0(U235) + k_1(U235)BU + k_2(U235)BU^2 \\ \rho(Pu) &= k_1(Pu)BU + k_2(Pu)BU^2 \end{aligned} \qquad (3)$$

where the k-s are calibration constants. The k-s were determined by fitting second order polynomials to the data from [18] corresponding to VVER-440 assemblies of 3.6 % initial ^{235}U enrichment. The obtained values are summarized in Table 1.

The burn-up, BU, was determined from the Cs activity ratio, as described in section 4. The concentration profiles are analogous to the measured burn-up profile shown in Fig. 5.

Table 1. The values of the parameters in the correlations between the burn-up and concentration of nuclear material given in % of the original uranium mass, obtained from the data in the DIQ of the Paks NPP [18] corresponding to VVER-440 assemblies of 3.6 % initial ^{235}U enrichment

	U-total	U-235	Pu
k_0	100	3,6	0
k_1	-1,50E-1	-1,06E-1	4,56E-2
k_2	4,24E-4	9,22E-4	-4,32E-4

5.2. Determining the mass of spent fuel

As mentioned above, "spent-fuel mass" is the sum of the masses of the remaining uranium, fission products and minor actinides. In this work we assumed that the mass of ^{137}Cs, *m(Cs137*) is proportional to the total mass of spent fuel as well as to the burn-up, *BU*, of the fuel. That is

$$m(Cs137) \sim m_{fuel} BU \ . \tag{4}$$

Therefore, the mass of spent fuel can be expressed as

$$m_{fuel} = C \frac{m(Cs137)}{BU} \ , \tag{5}$$

where C is some constant.

The intensity of radiation emitted by any particular fission product (in this case ^{137}Cs) in spent fuel is proportional to the mass of that fission product. However, the intensity of radiation registered by a detector is different form the intensity of emitted radiation, because not all emitted particles are registered. Some of the particles just pass by or through the detector, while others do not even reach the detector because of absorption by the source itself and by the materials between the detector and the investigated material. That is why the registered counts have to be corrected for detector efficiency and absorption in the intermediate materials, i.e. the detector has to be calibrated.

In our case the detected radiation is represented by the peak area *S(662)* of the 662 keV peak of ^{137}Cs, while the burn-up is represented by the activity ratio ^{134}Cs/^{137}Cs, also denoted above as *R*. Therefore, the detector signal which has to be calibrated against known spent fuel mass is *S(662)/R*.

In particular, we used the following relationship to determine the mass, m_{fuel}, of spent fuel within each canister with damaged fuel:

$$m_{fuel} = K \frac{S(662)}{R} \frac{1}{N} \tag{6}$$

where *K* is a calibration factor explained below, *N* is the number of sides from which an object was measured (generally *N*=3 for canisters and *N*=4 for assemblies), *S(662)* is the peak area of the 662 keV peak of ^{137}Cs evaluated from the sum spectrum $S_{canister}(i, v)$ obtained by summing all the spectra for the j-th canister in a given scanning direction v (up or down), while *R* is the cooling-time-corrected activity ratio ^{134}Cs/^{137}Cs evaluated from the same spectrum. The ratio R is proportional to the apparent (or effective) burn-up of the spent-fuel in the entire canister. In most cases it is close to, but **not equal** to the average burn-up.

In a similar way, the spent-fuel mass profile along each canister was constructed using the analogous formula

$$m_{fuel,i} = K \frac{S_i(662)}{R_i} \frac{1}{N} \tag{7}$$

where $S_i(662)$ and R_i are the respective values evaluated at the *i*-th scanning height, and the calibration factor *K* is assumed to be the same as in eq (6). This assumption was verified experimentally, by proving by measurement that $m_{fuel} = \sum_i m_{fuel,i}$.

Note that instead of using the count rate of the 662 keV peak, we used the peak area (divided by the number of sides from which the object was assayed) to determine spent-fuel mass. Since the scanning time was variable, but the scanning speed was constant for all objects, the peak area (and

not the count rate) in the summed spectrum is the quantity which is proportional to the total mass that has passed in front of the detector.

The calibration factor *K* was determined through calibration with regular spent-fuel assemblies of known initial uranium mass, and therefore, of known spent fuel mass. The calibration factor *K* consists of several parts as given below:

$$K = K_{Fe} K_{other} \frac{m_{Ufresh}}{\frac{S_{as}(662)}{R_{as}}} \tag{8}$$

where $m_{U,fresh}$ is the mass of total uranium in fresh fuel assemblies, $S_{as}(662)$ is the peak area of the 662 keV peak of ^{137}Cs evaluated from the sum spectrum obtained by summing all the spectra for a reference assembly in a given scanning direction (up or down), R_{as} is the activity ratio ^{134}Cs/^{137}Cs evaluated from the same spectrum, K_{Fe} is the ratio of the transmissions in the canister and assembly walls, given by

$$K_{Fe} = \frac{\exp(-\mu d)}{\exp(-\mu_{as} d_{as})} \tag{9}$$

where μ and d are the linear absorption coefficient at 662 keV and thickness of the canister wall, respectively, and μ_{as} and d_{as} are the same quantities for the reference assemblies ($\mu \approx \mu_{as} = \mu_{Zr} \approx \mu_{Fe} = 0{,}578$ 1/cm) and K_{other} is the ratio of the transmissions in all other materials (water, spent fuel) inside the canisters and assemblies.

An essential element of our approach to determining the spent-fuel mass is the assumption that $K_{other}=1$. This is a crude approximation, because the geometry of the fuel inside a canister is, in general, different from the geometry of the fuel in a regular spent fuel assembly, causing different absorption in the water and differences in self absorption. Nevertheless, measurements have proven that this approximation is justified, because in most cases the two opposite effects cancel out (more absorption in the water for smaller canisters goes together with less self absorption and vice versa for larger canisters), leaving $K_{other}=1$.

6. Results of the gamma spectrometric measurements

The mass of ^{235}U, total U and total Pu in each damaged-fuel canister was measured by gamma spectrometry.

By summing up all the data from the measurements of the individual canisters one arrives to the total mass of nuclear material in 72 canisters. The results in Table 2 were calculated from the summed spectra $S_{canister}(i, v)$ corresponding to each canister. The results for the total mass in 72 canisters obtained by summing up the mass-profiles constructed from the spectra $S_{300}(i, j, v)$ and the results obtained from the two spectra $S_{total}(down)$ and $S_{total}(up)$ agree with the results in Table 2 within the given uncertainty.

Table 2. The total nuclear-material content of the 72 canisters (average of the up- and downward scanning results)

Total U (kg)	^{235}U (kg)	Total Pu (kg)
3666 (± 10 %)	75 (± 15 %)	26 (± 20 %)

The ranges of applicability of the method are quite wide, as it can be seen from Table 3, which gives the largest and the smallest measured values for the mass of nuclear material.

Table 3. The smallest and largest measured values for individual canisters

	Total U (kg)	^{235}U (kg)	Total Pu (kg)
smallest measured value	7	0.13	0.05
largest measured value	121	3	1

Examples of the spent-fuel mass profile and apparent burn-up along a canister are given in Fig. 4 and Fig. 5 respectively.

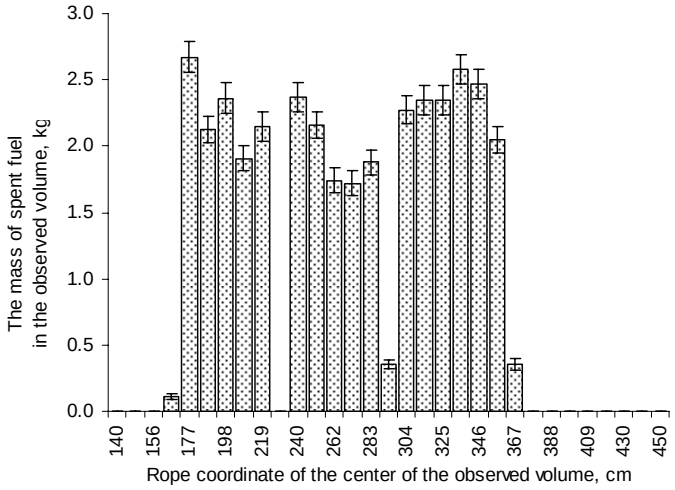


Fig. 4. The spent-fuel mass profile for the canister with identifier T29-025

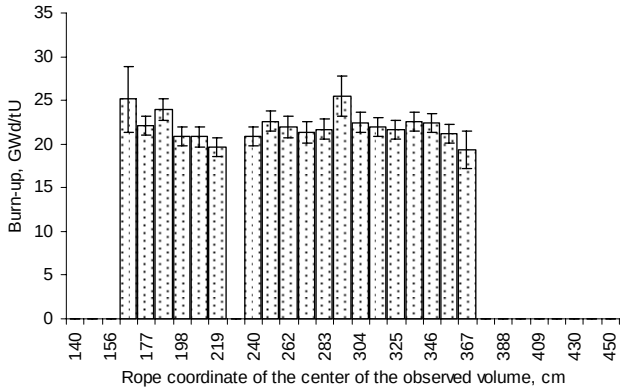


Fig. 5. The (apparent) burn-up profile of the canister with identifier T29-025

For the purpose of declaring the nuclear material-content to EURATOM and IAEA, in order to eliminate systematic errors, the measured values were normalized to the total mass to be accounted for. The detailed results and description of the measurements were handed over to the power plant, which, based on these results, declared the nuclear-material content of the canisters to EURATOM and IAEA. The declarations were accepted by both authorities.

7. Uncertainty

7.1. Estimating random errors: cross-validation with weight measurements

In order to increase confidence in the results of the gamma-spectrometric measurements, the data from measurements with other techniques were used. In particular, in parallel with the gamma spectrometric measurements, the neutron count rate, gross gamma count rate, and medium resolution gamma spectra were registered along each canister, as mentioned in section 3.1.

In addition the weight of material loaded into each canister was measured very accurately. From these data an estimate for the spent-fuel mass within each canister was given, by assuming that the density of material in each canister corresponds to the average density of the material in normal spent-fuel assemblies and that the ratio of spent fuel mass to the mass of other material (e.g. fuel cladding, broken construction elements) in the canisters is the same as for normal spent fuel assemblies [6].

To illustrate the uncertainties of the gamma-spectrometric measurements we use the weight measurements. The comparison of gamma spectrometry to the other measurement techniques will be presented elsewhere.

The sources of random error could be, e.g., the following

- uncertainty of self absorption coming from not knowing the exact geometrical position of the spent fuel within the canister
- using the apparent (effective) burn-up
- canister positioning uncertainty
- uncertainty of the irradiation history of the examined material
- the statistical error of the measurement device

Some of the sources of uncertainty mentioned above are negligible, while there could be also such sources of uncertainties which were not listed above. Therefore, instead of estimating the uncertainty contribution from each source to the uncertainty budget, here we estimate the random errors of the measurement of the spent-fuel mass based on cross-validation with weight measurements.

Consider the ratio $Q(i)$ of spent-fuel mass, $m_{fuel}(i)$, in the i-th canister obtained by gamma spectrometry, to the mass, $m_w(i)$, of material within the i-th canister obtained by weighing the canisters and assuming average density. Then we have

$$Q(i) := \frac{m_{fuel}(i)}{m_w(i)} = k_{Fe} k_{other} \frac{\frac{S(662)(i)}{R(i)}}{\frac{S_{as}(662)}{R_{as}}} m_{Ufreshz} \frac{1}{m_w(i)} = const \times \frac{\frac{S(662)(i)}{R(i)}}{m_w(i)} \quad (10)$$

where $m_w(i)$ is the mass of material in the i-th canister (including also fuel cladding) obtained from weight measurements.

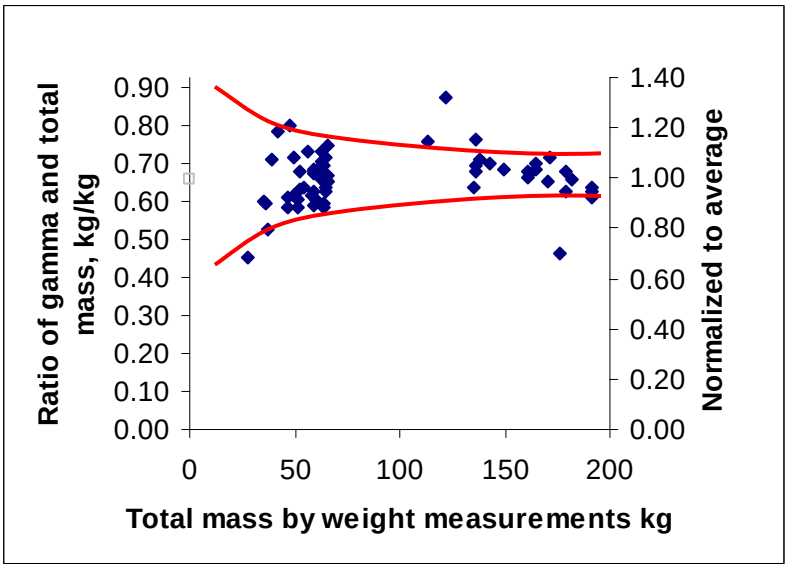


Fig. 6. The ratio of spent-fuel mass measured by gamma spectrometry to the total mass of material in the individual canisters, as a function of the total mass

In Fig. 6 one can see that the ratio *Q* is roughly constant, which means that its dispersion around its mean value is ("1 sigma") is, in general, less than 10 %. More precisely, for the canisters containing relatively intact parts of the assemblies (larger masses) the value of *Q* has a standard deviation of about 5 %, while for the canisters containing broken fuel pins and pellets (smaller masses) the deviation is about 10-20 % and it is increasing towards lower masses.

The constancy of *Q* means that

1. the total mass of material within a canister is proportional to the mass of spent fuel in that canister (justifying the use of the average density in the calculations based on weight measurements)
2. for each canister the "detector signal", that is, the quantity *S(662)/R* is proportional to the total mass of material loaded into that canister and to the mass of spent fuel (justifying the assumption that the absorption is the same for all canisters and assemblies, i.e. $k_{other}=const$)

Therefore, the dispersion of the quantity *Q* reflects at the same time the accuracy of both the gamma measurements and the weight measurements. Using two complementary measurement techniques, it was possible to justify the assumptions used in both approaches. Without the gamma measurements it would not have been possible to check to what extent is the assumption of "average density" correct, and without weight measurements it would have been much more complicated and less reliable to determine the total uncertainty budget of the gamma measurements

In Fig. 6 one can see at large mass values an outlier value for *Q* which is well below the others. It turned out that in this canister a damaged follower (control rod) assembly was placed. This assembly operated only for 1 cycle in the reactor, and within that cycle the lower part of the follower was never introduced inside the reactor core. Therefore, there was practically no spent fuel in that part the assembly, so no nuclear material was detected by our method which relies on measuring radiation from a fission product (^{137}Cs). However, the lower part of the assembly contained fresh nuclear fuel, that is, un-irradiated uranium. Therefore, in this case the results given by our method are lower than the true nuclear-material content of this particular canister.

There is another outlier in Fig. 6, which is well above the other values. By observing the 662 keV profile of the canister from the 3 different sides, we noticed that the asymmetry of the 3 sides is much larger than for other canisters. Furthermore, from the spent-fuel mass profile of the canister it was clear that on one part of the assembly there is much less material per unit length, as in other canisters. The power plant confirmed that parts of broken fuel pins fell out from the assembly, while it was being loaded to the canister. Less material implies less self-absorption, than used in our calibration. Therefore in this case the gamma spectrometric method overestimated the true nuclear-material content of the canister.

7.2. Reproducibility of the gamma-spectrometric measurements

The reproducibility of the measurements was verified by comparing the results obtained from the downward and upward scanning and also by repeated measurements of reference assemblies. (For one reference assembly also the detailed azimuthal profile was taken two times in 5 degree steps). The ratio of the upward and downward scanning results for the canister oscillates with a standard deviation of 3% around 1, indicating that the measurements are reproducible to a great accuracy.

8. Conclusion

The gamma-spectrometric method presented here proved to be the most accurate from all the available measurement options for determining the mass of nuclear material in damaged spent fuel. The accuracy of the measurements (~10 % on average for the total spent-fuel mass) is satisfactory from the point of view of safeguards requirements, and the operators declaration of its nuclear-material inventory was prepared based on the results of these measurements. The declaration was accepted by EURATOM and IAEA.

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[18] Design Information Questionnaire (DIQ) for the Paks NPP

Helping safeguards duties and control with a DVD

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Abstract:

The safeguard duties and controls are based on many National laws, implementing International agreement, treaty convention and protocols, and European directives.
The law 332 issued the 31 Oct 2003, implementing the Additional Protocol in Italy sees ENEA engaged as a technical advisor for the Department of Economical Development with specific studies and analysis and to help operators and authority control to satisfy the additional protocol requirements.
After the first collaboration with the inspectors from the IAEA, EURATOM and ISPRA (the Italian nuclear technical authority) it was decided to develop a software to help everybody involved in the additional protocol and dual-use items limited to category zero as defined in the European Directives.
Based on commercial software, all Italian laws are navigable on a DVD and from the additional protocol requirements there are direct links to specific pages of document, prepared by the U.S. Department of Energy, Nuclear Transfer and Supplier Policy Division, titled “Handbook for notification of Exports to Iraq – ANNEX 3 – United Nations Security Council Resolution (1996)” illustrating the materials and components with pictures and detailed description.
After the end of the technical consultation, it is possible to fill the format required by law and send it electronically to the competent authority, using the specific software developed by the European Commission in case of the EURATOM declaration.
The paper will show how the software works and useful help that can be obtained, taking to account that sometimes industry doesn’t know a real possible use of same material component and custom officers are facing even more difficulties.

Keywords: safeguards, additional protocol, dual-use

1. Introduction

The first step to implement the Additional Protocol was performed in compliance with the European Agreement requiring the use of a specific software for the “nuclear site declaration” and a formal procedure among ISPRA (Institute for Environmental Protection and Research) , European Commission and IAEA. In this first phase it was performed a campaign to identify the Italian nuclear sites, defined by the Additional Protocol in a broader ways compared with the past time, and advise them on the use of the specific software for the “declaration”.
The second step was to identify operators involved in the activities specified in the Annex I or producing materials or components described in the Annex II before the publication of the legal procedure to send relevant information to IAEA. This step was rather complex since it was the first time that non nuclear activities and materials are subject to “nuclear” legislation and the directive on dual-use material and components has to be applied too.
It was during this second step that the Department of Economical Development asked ENEA to help Italian operators with a specific software useful to know what are the activities to be declared or authorized and then to prepare the necessary documents.

ENEA in agreement with the duties established by law and by a contract with the Department of Economical Development has developed a specific software using Italian language as much as possible.

2. DVD content

The starting point was the software CAPE (Commission's Additional Protocol Editor) that was developed by the EC to manage in an efficient way all information related to the Additional Protocol collected through European Countries. The help manual of this software was translated in Italian language using commercial software to keep both the original English version and the Italian version.
In the first phase of the implementation of the Additional Protocol, ENEA performed a campaign to help nuclear operators to install and use the software CAPE, although the first declaration was performed by ENEA on behalf of the Department for Economic Development since it was impossible to appoint the site representative for the lack of specific legislation.
With the issue of the law the Additional Protocol was implemented through the site representatives very easily but some difficulties were found with industrial operators not aware about the new requirements fixed in the Annexes of the Additional Protocol.
Since most of the nuclear components and materials in the Annexes of the Additional Protocol are controlled by the dual-use directive too, ENEA was asked to prepare a new software to help operators to verify the requirements foreseen for their activities from the two legislations.
The new software starts as shown on fig.1 and a short guide gives advice to use it and the meaning of different colors used.

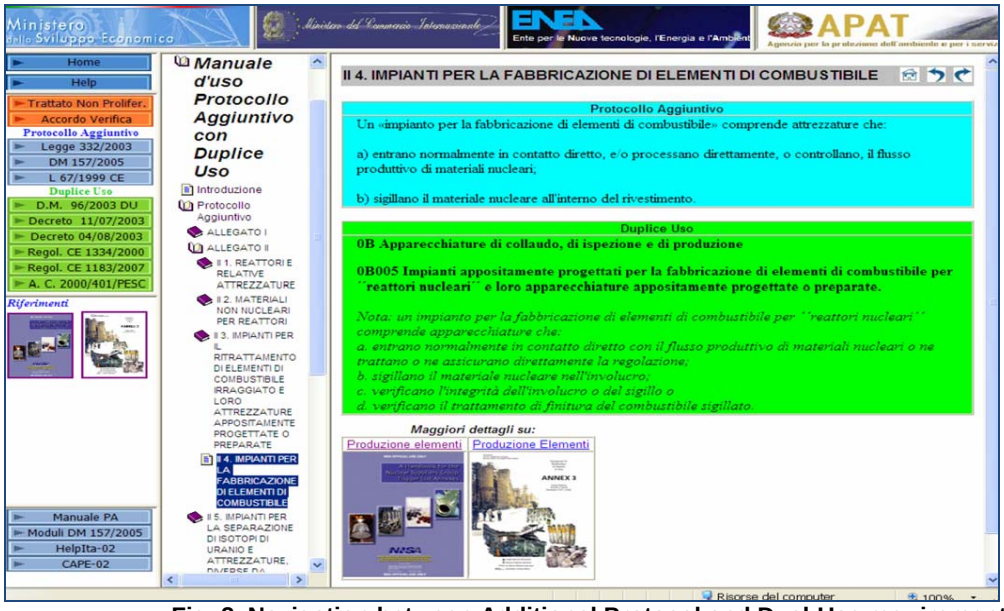
Fig. 1 – Starting page of the DVD content

The following documents are listed on the left of fig. 1 and marked with by different colors any times their content is shown on the screen (fig.2):

- IAEA INFCIRC/140 - Treaty on the Non-Proliferation of Nuclear Weapons (NPT)
- IAEA INFCIRC/193 - Verification Agreement between the IAEA and the European Atomic Energy Community (EURATOM)
- Commission Regulation (Euratoms) No 302/2005 of 8 february 2005 on the application of Euratom Safeguards

- Additional Protocol to IAEA INFCIRC/193 – 1999/188/Euratom
- Council Regulation (EC) No 1334/2000 of 22 June 2000 setting up a Community regime for the control of exports of dual-use items and technology
- Council Regulation (EC) No 1183/2007 of 18 September 2007 amending and updating Regulation (EC) No 1334/2000
- Council Joint Action of 22 June 2000 concerning the control of technical assistance related to certain military end-uses
- Council Decision of 22 June 2000 repealing Decision 94/942/CFSP on the joint action concerning the control of exports of dual-use goods
- Italian law n. 332/2003 - Ratification and execution of Additional Protocol to IAEA INFCIRC/193
- Italian Legislation Decree 9 April 2003, n. 96 Implementation of some disposition of EC Regulation n. 1334/2000
- Ministry of Economic Development Decree 11 July 2003 Institution of National Board for thr exportation of dual-use goods
- Ministry of Economic Development Decree 4 August 2003 Individuation of goods and countries of destination related to the exportation of dual-use goods as specified in regulation 1334/2000

Only the reference documents “A handbook for the nuclear supplier group Trigger list Annexes” and “Handbook for notification of export to Iraq – annex 3” are in the original language.



The blue background is for Additional Protocol related Material

The green background is for Dual Use related Material

Fig. 2 Navigation between Additional Protocol and Dual-Use requirements.

The operator can navigate throughout the documents and the Additional Protocol requirements are shown below the Dual-Use ones and more details are linked on the relevant pages of the reference documents: searching with keywords is available too.
Once the operator has checked what declaration has to compile, he can use the software CAPE, included and linked in the DVD, to prepare and send it electronically or, in the case of dual-use, what authorization he needs.

During the last meeting with Italian operators and control authorities organized by ENEA at Casaccia centre the 22nd May 2008, a copy of the DVD was distributed containing videos on the safeguards activities performed by IAEA.

Conclusions

Useful help can be obtained from using the DVD, taking into account that the Additional Protocol has enhanced the safeguards requirements on materials and components for the nuclear industry and

sometimes operators do not know a real possible dual-use of their products: custom officers are facing even more difficulties.
In the future, with the agreement of the Department for Economic Development, part of DVD will be on a web site and advice on safeguards duties and responsibility could be asked on line

Acknowledgements

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PERFORMANCE TESTING OF THE JSR-15 MULTIPLICITY SHIFT REGISTER AT HIGH COUNTING RATE

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Abstract:

Neutron multiplicity counting is a well established non-destructive assay method widely applied to the quantification of impure Pu bearing materials as well as bulk oxide and MOX. In these cases the advantage of multiplicity counting over traditional coincidence counting is the added information introduced by the third measured parameter which allows an additional parameter to be treated as an unknown in the analysis. To exploit multiplicity counting to the fullest, special neutron counters of high, uniform (in space) and flat (in energy) efficiency, also with short die-away characteristics and small dead-time are needed. Perhaps care also needs to be taken over the preparation of the items to be measured. For new applications these considerations can be taken into account at the design stage. However, as the envelope of applications gets extended attention to these factors alone is not sufficient to ensure integrity and usability of the data because of the limitations of existing multiplicity shift register units. In particular an increased clock speed beyond the present 4MHz is needed along with multiplicity histograms which extend beyond the present 0-255 range if counting rates in the MHz range are to be recorded with fidelity.

In this work we describe the JSR-15, a new generation, high performance portable multiplicity shift register with a clock speed of 50MHz, histogram registration of 0-511, and with Fast Accidentals sampling (FA) capability. The form factor, interface and command set is briefly reviewed against the present norms. Bench testing to prove the design are discussed and experimental data gathered with the JSR-15 from multiplicity counting system using NDA2000 software are presented showing how the unit performs as expected in high counting rate regimes which would saturate existing modules. The multiplicity data acquired marks a significant improvement in shift register capability and the results presented are a world first in extending the technique to 0-511 bins.

Keywords: shift-register, neutron multiplicity, neutron coincidence, high rate

1. Introduction

Multiplicity counting using multiplicity shift register (MSR) methods is well established and widely applied to the quantification of impure Pu bearing materials as well as bulk oxide and MOX. The traditional method of extracting the correlated rates relies on using the difference multiplicity histogram. A review of NMC is given in reference [1] although the reader is cautioned that the mathematics derivation is imprecise in treating induced fission whereas the final equations are correct. The advantage of multiplicity counting over traditional coincidence counting is the added information introduced by the third measured parameter which allows an additional parameter to be treated as an unknown in the analysis. To exploit multiplicity counting to the fullest, special neutron counters of high, uniform (in space) and flat (in energy) efficiency, also with short die-away characteristics and small dead-time are needed. Perhaps care also needs to be taken over the preparation of the items to be measured. For new applications these considerations can be taken into account at the design stage. However, as the envelope of applications gets extended attention to these factors alone is not

sufficient to ensure integrity and usability of the data because of the limitations of existing multiplicity shift register units. In particular an increased clock speed beyond the present 4MHz is needed along with multiplicity histograms which extend beyond the present 0-255 range if counting rates in the MHz range are to be recorded with fidelity.

Los Alamos has recently developed, in collaboration with Canberra Industries under CRADA (cooperative research and development agreement) guidelines, a hand-held multiplicity register (HHMR/JSR-15) The new module is a perfect replacement for the JSR-14 in the low end counting regime (i.e. below 4 MHz) and an excellent candidate for high end counting regimes which would saturate existing modules. It is now available commercially [2]. It is designed for use with CANBERAA NDA2000 [3] and the general-purpose International Neutron Coincidence Counting NCC software package, INCC [4].

2. Description of the JSR-15

The JSR-15 MSR, shown in figure 1, is a portable, fully computer-controlled, neutron analyzer that functions in the Canberra 2150 Multiplicity Mode. It a is a specialized pulse counter used primarily to count neutron events originating in neutron detection instruments. While the counter can be used to count any TTL or differential input pulse train with 20 ns pulse pair resolution, its ability to record time correlated events and the multiplicity distributions of these events renders it suitable for counting neutron events in the nuclear fields of material safeguards, waste assay and process monitoring and control. Fast Accidentals sampling is implemented and deeper registers have been used to accommodate the high sustained rates without overflows. The JSR-15 is a battery operated hand held device allowing operators to carry the unit into measurement areas with ease. The JSR-15 uses a USB pipeline to communicate with the user's computer. It complies with the International Atomic Energy Association (IAEA) neutron coincidence counting requirements [4].

Figure 1: JSR-15 Hand Held Multiplicity Shift Register. There is a single shifter register input and two extra scalers. The JSR-15 weight is 1.7 kg and its dimensions are 254x203x32 mm

The module has two interface mechanisms: a front panel pushbutton/LCD display and software including the CANBERRA NDA 2000 and the International Atomic Energy Association (IAEA) accepted LANL software, INCC. The software interface is through a USB serial interface. The JSR-15 is fully compatible with the existing CANBERRA NDA 2000 and INCC software.
The JSR-15 has built-in batteries and can run for up to eight hours on a single charge. The instrument has both a built-in charging circuit and a battery monitor circuit. The state of the battery is reported on the front panel LCD display.
The JSR-15 contains a user-programmable High Voltage (HV) power supply (0-2000V with maximum current of 700 μA) and a 5V supply (500 mA nominal) that can provide bias and power to the user's detector assembly (see figure 2).

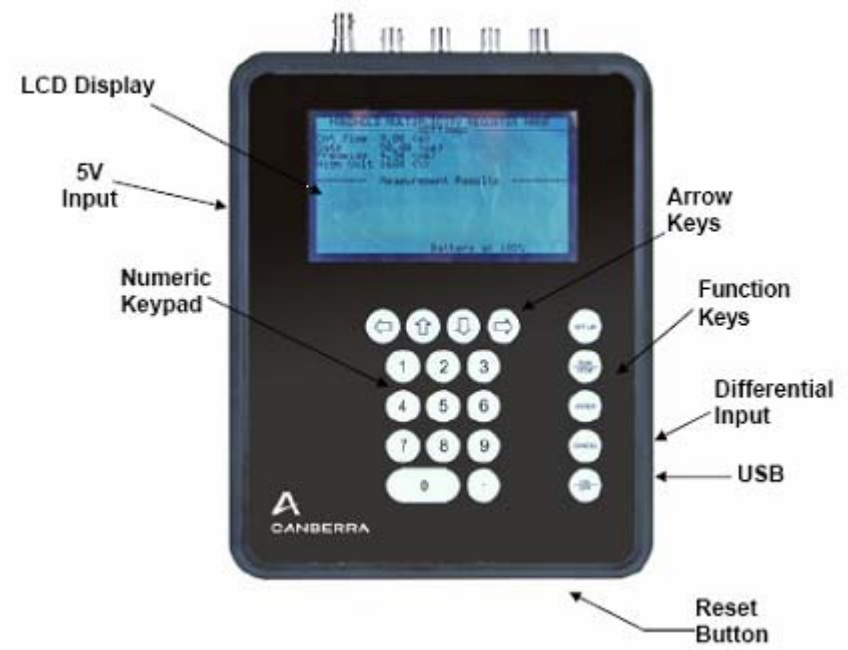

Figure 2: JSR-15 front panel display.

The connections to the JSR-15 are located on the sides and top of the unit. The main power input connection, 5 V, is located on the left side of the unit. The USB connection is located on the right side of the unit. It is a standard USB-B connector. The connections to the neutron measurement instrument are located on the top of the unit. There are three input channels, a 5V output and a High Voltage (HV) output. See Figure 3 for locations of these connectors and for the interconnection diagram.

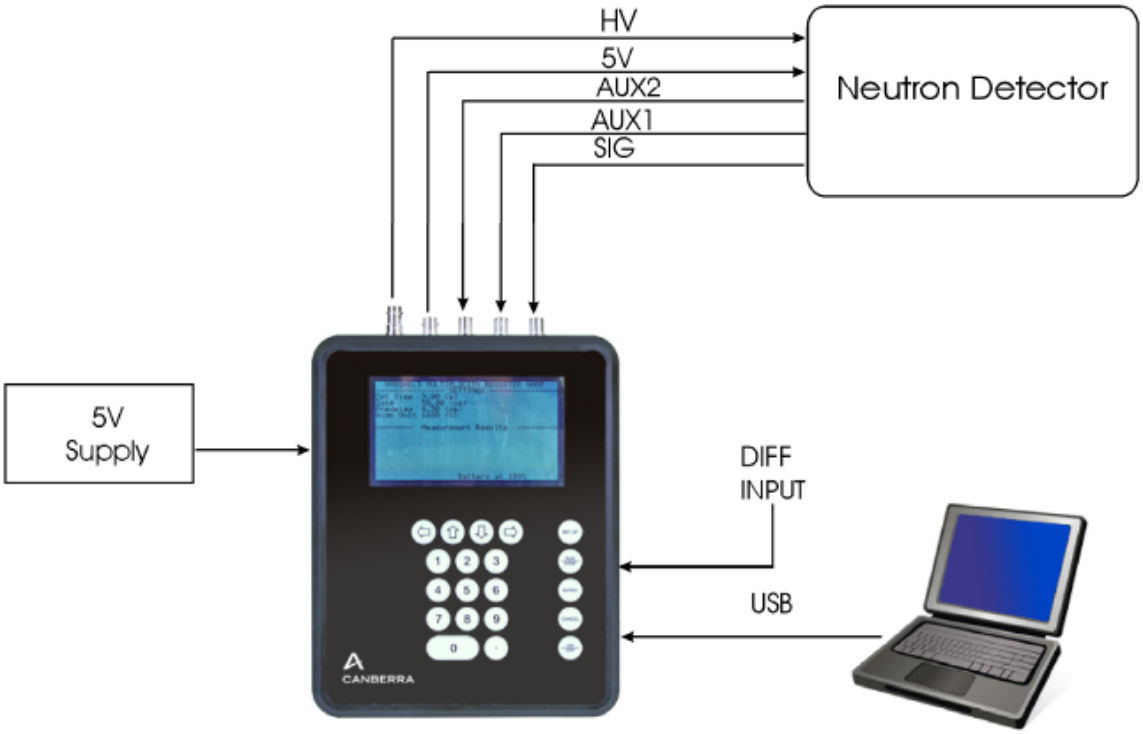

Figure 3: JSR-15 system interconnect schematics

2. Neutron counter

The high rate testing was performed on a Hexagonal Neutron Multiplicity Counting System (HNMS) with integral Gamma-Ray Isotopics measurement system (GIS) system for the assay of small containers of plutonium bearing products and residues. The section shown in figure 4 illustrates the He-3 tubes arrangements. The characteristics of the counters are described in Table 1.

Operating Parameter	Value
High Voltage	1780 V
Pre-delay time	2.5 μsec
Die-Away time	33.4 μsec
Gate Width	44 μsec
Dead-time Parameters (NCC)	a = 112.04 nsec b = 0 nsec2
Dead-time Parameter (multiplicity)	c = d =32.7 nsec
Dead-time parameter (δ)	31.6 nsec
Efficiency (Pu-240 point source)	0.420 ± 0.008
Efficiency (Cf-252 point source)	0.412 ± 0.008
Doubles Gate Fraction	0.6224 ± 0.0045

Table 1: HNMS characteristics. Parameters a, b, c, d are defined in [3]

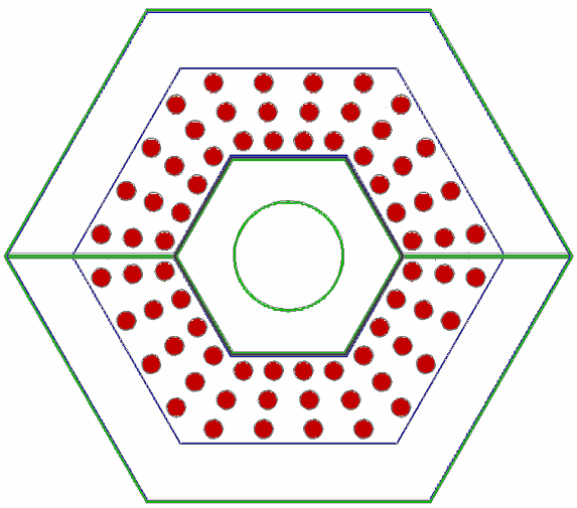

Figure 4: HNMS schematics

3. High counting rate testing

In a previous study [5] we showed that the JSR-15 is a perfect functional replacement of the JSR-14. The latter is intended only for normal (less than MHz) rates applications since it is limited by the 4MHz clock speed and multiplicity histograms which record only from 0-255 bins.
The tests were carried out using a series of Cf-252 sources. The measured Singles, Doubles, and Triples rates are shown in Table2. Notice that the present modules would saturates at 4 MHz for the Totals rate and at a rate less than that for Doubles and Triples. Table 3 shows that the JSR-15 performs as expected at high counting rate since the measured Ratios D/T, T/S, and T/D are flat across the wide dynamic range.

Sources IDs	**43825B (NPL)**	**43834B**	**43812B**	**43812B 43816B**	**43812B 43816B 43833B**	**43812B 43816B 43833B 43837B**
Singles [1/s]	21476.9	242713.9	1410773.2	2743473.9	4117825.9	5544727.7
	+/- 1.2	+/- 4.1	+/- 9.9	+/- 13.9	+/- 34.4	+/- 39.3
Doubles [1/s]	9572.9	108093.3	633841.5	1243214.2	1828889.9	2430835.2

	+/- 1.8	+/- 19.2	+/- 111.1	+/- 217.	+/- 655.2	+/- 867.2
Triples [1/s]	2738.9	30606.0	178875.6	362260.5	507872.5	711158.8
	+/- 2.0	+/- 76.1	+/- 1070.9	+/- 2922.6	+/- 10795.0	+/- 16583.7

Table 2: Measured Singles, Doubles, and Triples rates using JSR-15 and Cf-252 source after dead-time correction. The uncertainties are shown at ±1 standard deviation.

Sources IDs	**43825B (NPL)**	**43834B**	**43812B**	**43812B 43816B**	**43812B 43816B 43833B**	**43812B 43816B 43833B 43837B**
D/S	0.4457	0.4454	0.4493	0.4532	0.4441	0.4384
T/S	0.1275	0.1261	0.1268	0.1320	0.1233	0.1283
T/D	0.2861	0.2831	0.2822	0.2914	0.2777	0.2926

Table 3: Measured T/S, D/S, and T/D ratios using JSR-15 after dead-time correction

The excellent performance outlined in Table 2 is due to the increase in clock speed from 4MHz to 50 MHz and the JSR-15 capability to register multiplicity histograms between 0-511 bins. The multiplicity histograms for both the Reals+Accidentals and Accidentals distributions for the highest rate case (~5.5 MHz) is shown in Figure 5.

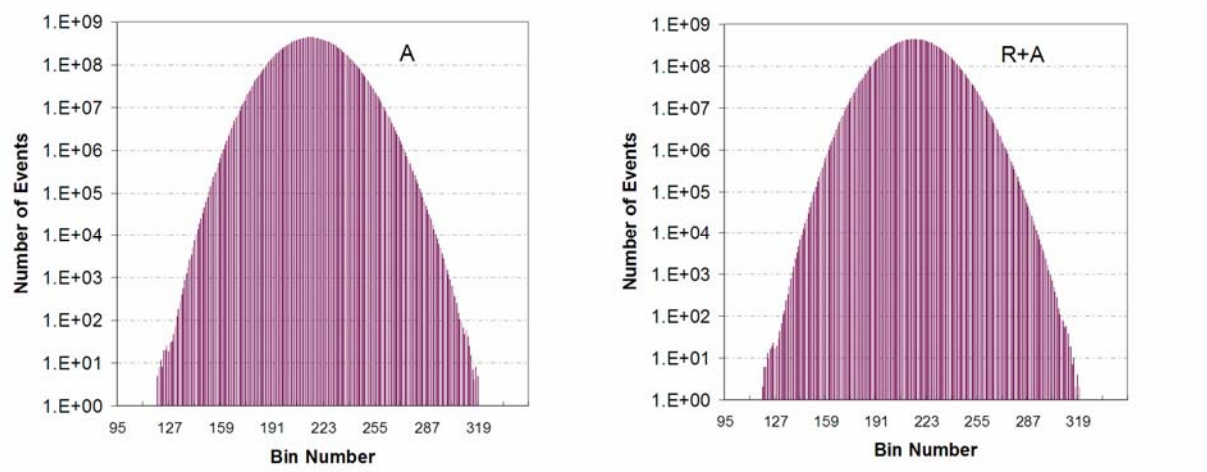

Figure 5: R+A and A multiplicity distributions using JSR-15 and the highest rate source listed in Table 2. The histograms are shown in Log scale. The acquisition time is 3600 sec.

4. Conclusions

Los Alamos has recently developed, in collaboration with Canberra Industries under CRADA (cooperative research and development agreement) guidelines, a hand-held multiplicity register (HHMR/JSR-15) that possesses a 50 MHz SR as well as an interactive front panel display with user settable parameters. The JSR-15 is a perfect replacement for the JSR-14 in the traditional low end counting regime (i.e. below 4 MHz) and an excellent module for high end counting regimes which would saturate existing modules. The multiplicity data acquired marks a significant improvement in shift register capability and the results presented are a world first in extending the technique to 0-511 bins. The JSR-15 is now commercially available.

8. References

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[5] N.Menaa, M. F. Villani, S. Croft, R. D. McElroy, R. Venkataraman, S. Philips and M. Newell; *Evaluation of the LANL Hand Held Multiplicity Register and Canberra JSR-15*, IEEE NSS/MIC 2007 proceedings

Compensating for Curium in wastes measured by neutron assay systems

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Abstract:

Passive neutron coincidence counting is a mature technique for assay of Pu in nuclear material, deployed in safeguards and waste inventory verification applications.

The presence of ^{242}Cm and ^{244}Cm in spent fuel wastes often poses a severe challenge owing to the short spontaneous fission half-life for these isotopes and the subsequent prolific spontaneous fission neutron emission. For most waste assay applications, neutron assay techniques are not capable of distinguishing between the spontaneous fission neutrons emitted by these Cm isotopes and ^{240}Pu, which is normally the principle nuclide of interest. Therefore, the presence of even small quantities of these isotopes can result in gross over-estimation of the Pu inventory, if an appropriate correction is not made.

In this paper we first describe the nature of the Curium challenge in waste assay applications. We review the basic nuclear data (half-lives and neutron multiplicity data) and illustrate the potential magnitude of the problem with reference to typical Curium levels in wastes. We then describe possible techniques for dealing with this problem, highlighting the benefits and limitations of each. These include neutron multiplicity measurements, complementary use of passive and active neutron interrogation, gamma spectrometry, and use of plant-supplied isotopic fingerprint data. Finally, we comment on how the optimum selection of assay equipment and algorithms at the design stage for a new facility can minimise future problems from Curium interference.

The use of multiplicity counting is not generally useful owing to the lack of differentiation between the multiplicity ratios for the even isotopes and ^{240}Pu when compared with the variations expected for real wastes. However, where defensible plant fingerprints available, simply allowing for the contribution of Curium to the ^{240}Pu effective response, can represent a powerful solution to avoid over-pessimistic over-estimation of the Pu inventory. When fingerprints cannot be relied upon and a direct fissile measurement (Pu mass) is required, active neutron techniques are often used to overcome the high passive neutron background from Curium and other isotopes and for such applications it is vital to select the correct measurement technology at the earliest stage in the project lifecycle.

Keywords: Curium, neutron detection, Plutonium detection, neutron coincidence, neutron multiplicity

1. Introduction

Passive neutron coincidence counting (PNCC) is a widely used and mature technique for the assay of Pu in nuclear material, deployed in safeguards and waste inventory sentencing and verification applications. Pu is quantified by measurement of pairs of spontaneous fission neutrons from principally the even isotopes of Pu.

The presence of ^{242}Cm and ^{244}Cm in spent fuel often poses a severe challenge to the assay of Pu owing to the short spontaneous fission half-life (much shorter than for ^{240}Pu) for these isotopes and the subsequent prolific spontaneous fission neutron emission. Most neutron assay techniques are not capable of distinguishing between the spontaneous fission neutrons emitted by these Cm isotopes and ^{240}Pu, which is normally the principle nuclide of interest. Therefore, the presence of even small quantities of these isotopes in spent fuel residues can result in gross over-estimation of the Pu inventory, if an appropriate correction is not made. The magnitude and significance of the potential problem depends upon the specific goals of the assay application, and the nature of the processes that could have given rise to the Curium – contaminated material. Cm in Spent Nuclear Fuel can be used to mask Pu diversion and also expands the measurement uncertainty. It is therefore important to quantify its impact.

In this paper we review the basic nuclear data (half-lives and neutron multiplicity data) and illustrate the potential magnitude of the problem with reference to typical Curium levels in wastes. We then describe possible techniques for dealing with this problem, highlighting the benefits and limitations of each. These include neutron multiplicity measurements, complementary use of passive and active neutron interrogation, gamma spectrometry, and use of plant-supplied isotopic fingerprint data. Finally, we comment on how the optimum selection of assay equipment and algorithms at the design stage for a new facility can minimise future problems from Curium interference.

2. Nuclear data

Plutonium, Americium and higher actinides (including ^{242}Cm and ^{244}Cm) are produced in nuclear reactor fuel as a result of successive neutron capture and beta decay. In a given nuclear reactor heavier isotopes require a greater number of neutron capture cycles and so require a longer period of irradiation to form in significant quantities. The concentrations of ^{242}Cm and ^{244}Cm depend upon the fuel burnup (the length of time the fuel was present in the reactor), the neutron flux and the fuel position within the reactor. The fewer neutron captures required to form ^{242}Cm means that this isotope is formed initially, in greater concentrations than ^{244}Cm. Intense irradiation in reactors can produce higher isotopes such as ^{250}Cf and ^{252}Cf which are also important spontaneous fission neutron sources. However, production of these sources usually requires special high neutron flux irradiation conditions (and targets) and these isotopes are not generally important in typical fuel residue / waste assay applications. Figure 1 illustrates the principle paths by which ^{242}Cm and ^{244}Cm are formed by irradiation of Uranium.

The half-life for ^{242}Cm is 163 days whilst the half-life for ^{244}Cm is 18.1 years. For fuel with a short cooling time ^{242}Cm can still be present in large quantities and dominate the neutron emission. However for aged fuel which has been stored for many years after removal from the reactor, or waste arisings in legacy plant undergoing decommissioning, any ^{242}Cm has usually decayed to negligible levels such that ^{244}Cm dominates.

We now consider the basic nuclear data for ^{242}Cm and ^{244}Cm, in the context of conventional Passive Neutron Multiplicity Counting (PNMC). Neither of these isotopes has a significant thermal neutron fission cross-section and so they are not relevant in consideration of the response of active neutron interrogation systems (used to quantify total fissile mass). Considering dispersed material with negligible self-multiplication, the simplified point model equations for the Totals (T), Doubles (R_2) and Triples (R_3) rates per unit mass of a spontaneously fissionable isotope, are given by equations 1, 2 and 3 respectively.

$$T = g.\varepsilon.\nu_t(1+\alpha) \quad \text{Eq 1}$$

$$R_2 = g.\varepsilon^2.f.\frac{\nu_2}{2!} \quad \text{Eq 2}$$

$$R_3 = g.\varepsilon^3.f.\frac{\nu_3}{3!} \quad \text{Eq 3}$$

where

g is the spontaneous fission rate (fissions per second per gram),
ε is the detection efficiency,
f is the coincidence gate utilisation factor,
ν_t is the spontaneous fission total prompt neutron yield per fission,
ν_2 is the second factorial moment of the spontaneous fission prompt neutron multiplicity distribution,
ν_3 is the third factorial moment of the spontaneous fission prompt neutron multiplicity distribution,
α is the ratio of (α,n) to spontaneous fission neutron production.

In equations 1, 2 and 3, we note that the terms ε and f are, to first order, independent of the spontaneously fissionable isotope. The efficiency, ε, can vary from item to item. Cm and Pu may be but are not always co-located and so ε may but may not be nearly equal for the two. Matrix and spatial variation leads to an uncertainty in ε which is typically much larger for wastes than for fuel assay safeguards counters. The count rate per unit mass therefore depends on the values of g and ν_t (for Totals) ν_2 (for Doubles) and ν_3 (for Triples). This nuclear data has been previously evaluated, and is summarised in Table 1 for the most important spontaneously fissionable isotopes in safeguards applications. This data has been extracted from references [1] and [2], the raw data being based mainly on evaluations by Zucker and Holden [3, 4 and 5].

Isotope	Half life $\tau_{½}$	Spontaneous fission branching ratio (%)	g (fiss.s^{-1}.g^{-1})	ν_t	ν_2	ν_3
240Pu	6564 y	5.7E-06	4.75E+02 (1.6%)	2.140 (5)	3.789 (2)	5.11
238Pu	87.7 y	1.9E-07	1.17E+03 (1.9%)	2.21 (8)	3.957	
242Pu	3.733E+05 y	5.5E-04	8.07E+02 (1.0%)	2.156 (14)	3.809 (36)	
242Cm	163 d	6.2E-06	7.81E+06 (2.9%)	2.54 (2)	5.13 (1)	8.04
244Cm	18.1 y	1.3E-04	4.11E+06 (1.5%)	2.72 (2)	5.94 (9)	10.1
238U	4.468E+09 y	5.0E-05	6.78E-03 (1.2%)	1.99 (3)	2.874 (141)	

Table 1. Nuclear data for ^{242}Cm and ^{244}Cm and other important spontaneously fissionable isotopes. Uncertainties on the last significant figures are shown (where available) in brackets (% uncertainties for g data). The values for g, ν_t and ν_2 were obtaiend from [1]. The values for ν_3 were obtained from [2].

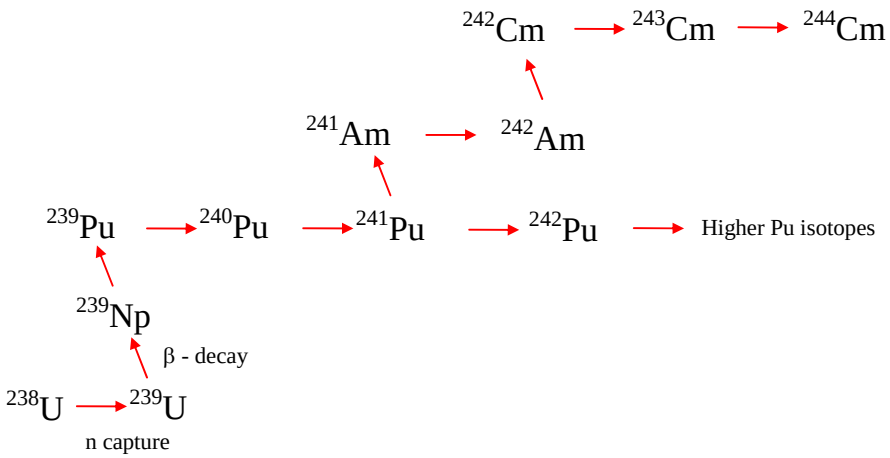

Fig. 1 Routes for production of ^{242}Cm and ^{244}Cm. This could be extended to show production of other (less important due to the low concentrations) spontaneously fissionable isotopes such as ^{246,248}Cm and ^{250,252}Cf.

The result of a PNCC measurement is typically expressed in terms of a ^{240}Pu effective mass, m_{eff}. The result will contain contributions from the even isotopes of Pu, as well as ^{242}Cm and ^{244}Cm. As we have noted previously, in conventional PNCC counting it is not possible to discriminate between the contributions from these isotopes. To illustrate the domination of Curium even at low relative abundance, equation 4 expresses the ^{240}Pu effective mass in terms of the contributions from the various isotopes. This equation shows that one gram of ^{244}Cm gives a doubles signal equivalent to 13.551 kg of ^{240}Pu.

$$m_{eff} = m\left(^{240}Pu\right) + 2.572m\left(^{238}Pu\right) + 1.708m\left(^{242}Pu\right) + 22250m\left(^{242}Cm\right) + 13551m\left(^{244}Cm\right) \qquad \text{Eq 4}$$

3. Impact of Curium in waste assay

The intense neutron emission from Curium is often exploited, for example in burnup measurements of spent fuel. Measurement of burnup in this manner relies on known correlations between the fuel burnup and the ^{244}Cm / ^{242}Cm activity concentrations. Spontaneous fission neutrons from these Curium isotopes is likely to dominate the neutron emission, such that total neutron counting (usually in a fission chamber or proportional counter) is sufficient (and preferred because exceptional neutron / gamma discrimination performance is required). Generally in waste assay Totals neutron counting is not used quantitatively because of the high and variable (α,n) rate. If Curium dominates, however, Passive Totals counting can become useful. Triples are usually statistically overwhelmed by Accidentals and so are not viable for quantitative analysis.

In waste assay applications, accurate and reliable knowledge of the waste "fingerprint" is not always available. This is common due to the variable and often uncertain nature of the processes that generate the waste. In support of decommissioning and waste management programs there is increasing emphasis on characterising specific waste streams according to the known nature of the activities that were prevalent in specific areas. However the radionuclide composition of waste in a particular container (for example a 200 litre drum) depends on the method of filling the drum and whether or not this could include wastes from different areas including cases which are subject to a variable quality of characterisation. If a project team has a high degree of confidence in the nuclide "fingerprint" for a specific waste stream, the activities of particular nuclides can be inferred by measurement of prominent "key" nuclides. On the other hand, when measurements are performed for sentencing or auditing purposes, it is important that the adopted assay methodology can deal appropriately with unexpected waste forms and nuclides including Curium. Thus the impact of Curium in wastes depends on the objectives and purpose of the measurement systems.

The presence of ^{244}Cm and / or ^{242}Cm can have the following impacts if appropriate compensation techniques are not considered:

- If PNCC measurements are performed based on an assumed Pu isotopics vector without consideration of the potential presence of ^{244}Cm / ^{242}Cm (either because testing for these isotopes was not considered as part of the process of determining the fingerprint or because it is believed that the waste stream is free of these isotopes in any case), then the unknown presence of these isotopes can result in severe over-estimation of the measured ^{240}Pu effective and hence the ^{total}Pu mass. This over-estimation can be an order of magnitude or more, depending on the isotopic composition of the material. The elevated neutron emission can lead to false categorisation of wastes and un-necessary rejection from waste treatment facilities based on the measured Pu alpha activity.

- Lack of accurate characterisation of the range of potential waste streams for the Curium content, leads to a large uncertainty in the response. Setting a calibration according to a conservatively high Curium content can result in potential under – reporting of the Pu content.

- In typical systems which use a combination of PNCC to measure the Pu mass and active neutron counting to measure the total fissile mass, the ^{239}Pu mass is obtained by combining the measured ^{240}Pu effective mass with the assumed Pu isotopics and the ^{235}U mass is obtained by subtracting the resulting contribution (from ^{239}Pu) to the total fissile mass

response.

The unknown presence of Curium can result in a false ^{240}Pu signal but the absence of a genuine fissile ^{239}Pu contribution to the total fissile response can give rise to a highly negative ^{235}U mass which poses challenges for the interpretation of the assay results.

- The presence of unsegregated waste streams with highly variable and unknown Curium content, will result in large uncertainties in the system calibration.

- The presence of ^{242}Cm / ^{244}Cm in significant quantities can act as an intense neutron background source which means that detection of a neutron signal from Pu is subject to a reduced quality of neutron counting statistics.

4. Techniques for Curium compensation

Having highlighted the potential impact of Curium for fissile material waste assay applications and noted the importance of recognising the objectives of the assay, we now explore the various physics – based compensation techniques. It is strongly recommended that subject matter experts are involved in the analysis of assay results exhibiting Curium problems, on a case by case basis.

4.1 Allowance for presence of Curium in Pu isotopic fingerprint

As we have seen, most PNCC – based Pu assay systems rely on a measurement of the ^{240}Pu effective mass, $m_{eff,}$ from the coincidence "Reals" or doubles response. A simple PNCC measurement cannot distinguish between the contributions to the response from different spontaneously fissionable isotopes. It is usual to extract the ^{total}Pu mass from the measured m_{eff}, by multiplying by the ^{total}Pu / ^{240}Pu effective mass ratio. This ratio is obtained from an assumed Pu isotopic fingerprint set (often selectable by the operator, matching fingerprints to the known origin of each waste container being assayed) in combination with equation 4. If one does not suspect Curium to be present in the waste streams, it is common practice for only the even Pu isotopes to be considered in equation 4. However, if one has implemented a robust waste characterisation procedure, it is possible to examine samples of each waste stream for the presence of ^{242}Cm and ^{244}Cm. If the results are deemed to be sufficiently accurate and representative of the actual bulk waste arisings, then the relative concentrations can be simply entered into the Pu isotopic fingerprint calculation (equation 4) as described above. The uncertainty in the isotopic composition based non assessment of the waste generation processes should be propagated along with other experimental uncertainties.

Various techniques are used to determine Pu / Cm relative concentrations. If wastes packaged into a bulk waste container can be reliably traced back to materials / fuels with a known history (reprocessing operations, fuel properties, etc), then the results of standard reactor physics codes (sometimes known as "burnup codes" or "depletion codes") can be used, of which the established codes ORIGEN [6] and FISPIN [7] are typical examples. Alternatively, it is possible to take small samples and perform laboratory analysis using standard analytical techniques such as alpha spectrometry. ^{242}Cm has a characteristically high energy α particle emission at 6.1 MeV while ^{244}Cm also has a high energy α particle emission at 5.8 MeV. These energies are significantly higher than the α particles from ^{241}Am and the Pu isotopes. Such characterisation is often performed for poorly characterised waste streams of uncertain origin. Care is obviously required to ensure that the resulting fingerprints are adequately representative of the bulk waste being measured by neutron counting. The sampling and destructive analysis support work needs careful planning and execution.

This technique has the advantage that it automatically allows for the known presence of the even Curium isotopes in waste streams, and thereby avoids the potential gross over-estimation that might occur in the Pu mass if it is assumed that there is no Curium present. It is suitable for applications where it is possible to obtain defensible waste stream characterisation including Pu and Cm isotopes. The principal disadvantage of this method is that it relies on robust characterisation procedures, and may be susceptible to under-reporting of the Pu mass if an incorrect fingerprint is selected.

4.2 Neutron Multiplicity Measurements

It is possible, under some conditions, to exploit the differing spontaneous fission neutron multiplicity distributions for the various isotopes. Table 1 compares the 2nd (doubles) and 3rd (triples) moments of the spontaneous fission neutron multiplicity distribution. The average multiplicities for ^{242}Cm and ^{244}Cm are significantly greater than for ^{240}Pu. The singles rate is not recommended for quantitative Pu or Cm assay, because it is greatly affected by (α, n) reactions whose yield depends critically on the chemical form of the Pu which is often highly uncertain (the yields can be increased greatly by the presence of even trace quantities of low – atomic number chemical impurities such as fluorine). The higher order multiplicities are not generally useful due to the low count rates, however the triples to doubles ratio can be diagnostic of the spontaneously fissionable species when both are statistically viable and certain conditions are met.

There are several ways in which PNMC can be exploited, either to provide a "warning flag" for the presence of Curium, or to provide quantitative Pu and Cm co-assay. Accuracy and reliability of such measurements can, however, only be assured from a thorough understanding of the instrument calibration, assumptions and performance (total measurement uncertainty, spatial response uncertainties, matrix variability, physical / chemical form of Pu / Cm and relative spatial distributions).

Inspection of Table 1 shows that the ν_3/ν_2 ratio is 1.37, 1.57 and 1.70 for ^{240}Pu, ^{242}Cm and ^{244}Cm respectively. Thus under ideal conditions [2], assuming that the Pu and Cm are co-located, the isotopic ratios are fixed, the efficiency is constant and known, and there is no self-multiplication and no enhancement of higher order multiplicities due to (α, n) reactions, it is possible to discriminate between ^{240}Pu and ^{244}Cm (and, less reliably, ^{242}Cm) by measuring the triples to doubles ratio.

However, such analyses are subject to the following serious disadvantages when applied to large containers which are normally used for waste assay:

- The triples counting rate is low, depending on the cube of the detection efficiency. This means that discrimination is only useful for relatively large quantities of Pu, well above the detection limit for typical assay systems.

- The variation of efficiency throughout a waste drum (unknown spatial location, uncertainty typically 10 – 20 % for "benign" matrices) can produce a Triples / doubles ratio variation which exceed the ν_3/ν_2 difference being sought. This effect is even more pronounced if the Cm and Pu are not co-located.

- If the perturbation of the matrix compared to the reference calibration (e.g. viable experimental probes) is not well known, the efficiency uncertainty can again exceed the ν_3/ν_2 ratio being sought.

- If one of the constituents (Cm or Pu) dominates the counting rate, the statistical accuracy with which the other can be determined, is poor.

Considering the above limitations, a useful implementation of PNMC can be to provide a simple warning "flag" for the presence of a problematic gross quantity of Cm. Evaluation of the appropriate triples / doubles ratio threshold setting relies on careful consideration of the above limitations, in order to avoid false indications for the presence of Cm.

An alternative approach is to implement separate calibrations for Pu and Cm, and solve the resulting simultaneous linear equations (contributions from both Pu and Cm) for the doubles and triples rates. This technique is subject to similar assumptions and limitations as described above, and is hence suited to relatively well characterised waste streams. As an example of incorporation of this technology into automated data acquisition / analysis software, Canberra's NDA2000 Non Destructive Assay Software [9] permits Cm and Pu co-assay using this method.

It is concluded that PNMC can be a useful technique under limited conditions where the waste streams are well understood and rather uniform, for example when calibration is based on known homogeneous matrices with a reliably uniform spatial distribution of Pu / Cm and where the assumption of co-location is reliable. For sophisticated waste assay systems which must deal with

highly variable and uncertain waste streams (for example, measuring poorly characterised legacy waste) which rely on matrix compensation and possibly algorithms to attempt to correct for spatially non – uniform Pu distributions (see, for example, reference [8]), PNMC measurements for Cm discrimination will be subject to large uncertainties and complex assumptions which is likely to lead to poor reliability. Interpretation of the results is then best handled as part of an expert review process in which all available information can be assessed together.

4.3 Gamma techniques

Gamma spectroscopy is often used in tandem with passive / active neutron measurements, for waste sentencing and verification measurements. Quantitative U and Pu assay can be performed under some conditions, although this is limited by the presence of the background gamma flux from fission and activation products). Commonly, High Resolution Gamma Spectrometry (HRGS) measurements are used to measure the Pu isotopic composition of waste, which is in turn used in combination with the ^{240}Pu effective mass measured by PNCC, to determine the ^{total}Pu mass. When applied to challenging waste assay problems, particularly poorly characterised legacy waste, HRGS can also be used to achieve a better understanding of the assumptions and limitations of quantitative PNCC assays. Such use of complementary assay technology is advisable due to the independence of the different techniques (different failure modes, assumptions, sources of uncertainty), leading to more robust conclusions.

In the context of Curium interference with the measurement of Pu by neutron counting, it is possible to measure the characteristic γ emission from ^{244}Cm. For example, by quantifying the characteristic 152.63 keV gamma emission from ^{244}Cm, it is possible to estimate the ^{244}Cm level. There is an interfering γ emission from ^{238}Pu however the contribution from this can be estimated by performing simultaneous Pu isotopics analysis with a code such as MGA or FRAM [10] and examining the results together with the PNCC ^{240}Pu effective result. The low γ energy means that attenuation will be high and uncertainties will be large especially for large containers with heterogeneous, heavy matrices. However, our studies show that this technique may, under some conditions and with careful assessment of the uncertainties through calibration and modelling, be used to place an upper limit on the concentration of ^{244}Cm. This may provide a “flag” for the presence of Curium, and when used quantitatively can be used to add to the robustness of a waste sentencing declaration. Gamma ray measurements are typically most useful in this context for contact handleable waste and we note recent work in making codes such as MGA and FRAM more robust for waste applications.

4.4 Use of passive and active neutron counting

As discussed earlier it is often necessary to consider the impact of the increased passive neutron emission (from Curium or other neutron emitters including random neutron (α, n) emission from low – atomic number elements such as fluorides), on the detection limit for Pu assay by passive neutron assay. This is important when one is attempting to measure the Pu spontaneous fission signal in the presence of an unknown and possibly variable neutron background from Curium. The achievable detection limit for Pu assay is elevated in the same way that the Passive mode Pu detection limit is increased due to the high neutron coincidence background from cosmic ray interactions in high atomic number materials (e.g. lead shielding in flasks). The result may be that it is impossible to measure Pu below an upper threshold which must be demonstrated for plant control and / or accountancy purposes. The (α, nγ) reaction lines (the γ ray energy is characteristic of the “target” isotope) can be flagged in the γ-ray spectrum and can be used in some cases to estimate the associated neutron production [11] but this approach is not widely used and is currently best left as part of the expert review process.

A common solution is to use active neutron interrogation instead, based on either californium shuffler or Differential Die-Away (DDA) technologies. In these techniques, an intense neutron source is thermalised within the assay chamber, and used to interrogate the waste sample. The induced fission neutrons from the fissile isotopes (^{239}Pu) are then counted to allow a direct measurement of the fissile mass. Known Pu isotopics can be used to determine the masses of the other isotopes if necessary. This technique is valuable due to the ability to provide a direct measurement of the fissile mass, which is normally of direct interest for safeguards and criticality control purposes. The detection limit can be improved by increasing the strength of the neutron source. Californium shuffler measurements offer detection limits of typically fractions of a gram of total fissile material, whereas DDA measurements

routinely achieve detection limits of a few – 10's of milligrams of total fissile material, for typical 200 litre waste drums.

In many waste assay chambers, the Pu and U masses are measured by combining the PNCC and active mode results. A PNCC measurement provides the ^{240}Pu effective mass from which (using assumed or HRGS measured Pu isotopics) the masses of the other Pu isotopes are derived, and a californium shuffler or DDA measurement provides the total fissile mass (containing contributions from principally ^{235}U and ^{239}Pu). Subtraction of the known ^{239}Pu response (from PNCC + known isotopics) then yields the ^{235}U mass.

The unknown presence of ^{242}Cm or ^{244}Cm will generate a false ^{240}Pu effective signal which is not reflected in an associated ^{239}Pu active mode response. Therefore the use of an assumed "Pu – only" isotopics isotopic set will result in subtraction of an excess ^{239}Pu signal with the possible result of a highly negative and confusing ^{235}U mass ! It is possible to look for such conditions which can only be caused by the presence of Curium, and use this as a "flag", suggesting that the passive response is unreliable and that the operator should rely instead on the active mode "total fissile" result. Such a procedure requires careful quality assurance and validation testing, in order to ensure correct interpretation of plant data.

Photofission, performed at several energies, provides an alternative active neutron interrogation technique but this is not one widely used in production / waste management plants at this time.

5. Conclusions

The presence of Curium waste streams gives an elevated spontaneous fission neutron emission which produces substantial problems in Pu assay applications. The nature of the problems is specific to the nature of the particular assay systems, depending on the ultimate purposes of the assay systems, how the results of the systems are used, the method of calibration, the assumptions and limitations of the analysis algorithms and also the amount of prior knowledge in the waste stream characteristics.

Passive Neutron Multiplicity Counting and gamma spectroscopy can be applied to provide a warning flag for the unexpected presence of Curium, thereby avoiding the problems of over-estimation of the Pu mass. However, these techniques are best suited to applications with homogenous, segregated wastes with a known spatial distribution of (co-located) Pu and Cm. Active counting can be used instead of passive neutron counting, for Pu assay applications where the neutron background from Curium could otherwise lead to a poor signal-to-background ratio such that it is not possible to meet the required Pu detection limits by passive neutron assay. The most powerful and accurate means of compensating for the presence of Curium is based on a calibration which allows for the known Curium concentration together in the Pu isotopic fingerprints. This is best achieved where wastes are carefully characterised and segregated *at source*, prior to bulk waste measurements in neutron assay chambers.

Some techniques, such as multiplicity counting and modification of Pu isotopic fingerprints to include Curium, can be readily implemented to most passive neutron multiplicity counters by software modification / recalibration. Appropriate use of active neutron assay technology depends on careful consideration of the project objectives at the design stage, determining the measurement performance (signal to background ratios and detection limit goals) will identify whether passive or active counting is appropriate. Since the assay chamber design features will differ substantially between passive and active counters, such studies are best performed early in the project lifecycle, to ensure that the plant design properly accommodates the assay chamber design requirements.

A key feature of the Curium compensation strategies discussed in this paper, is that prior characterisation of waste streams always improves the accuracy of fissile material assays and the robustness of the compensation. Early involvement of subject matter experts at all stages and also during data review is strongly recommended.

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European Union Satellite Centre

MONITORING NUCLEAR FACILITIES USING SATELLITE IMAGERY

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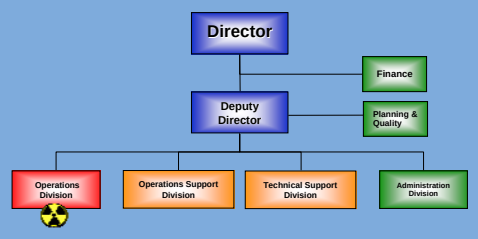
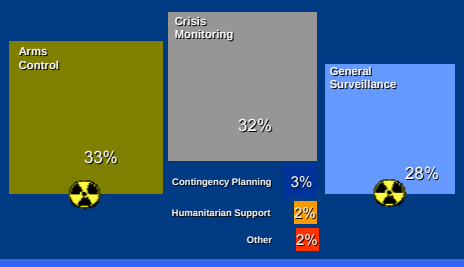
EUSC MISSIONS

The mission of the European Union Satellite Centre (EUSC) is to support the decision-making of the European Union by providing analysis of satellite imagery and collateral data. The EUSC is an Agency of the Council of the European Union. It is one of the key institutions of the European Security and Defence Policy (ESDP), and the only one in the field of space.
The EUSC delivers products derived from the analysis of satellite imagery and collateral data, and assures technical developments in direct support to its operational activities and specialised training for its own Imagery Analysts (IAs), as well as participants from EU Member States and Third States.

EUSC ORGANISATION



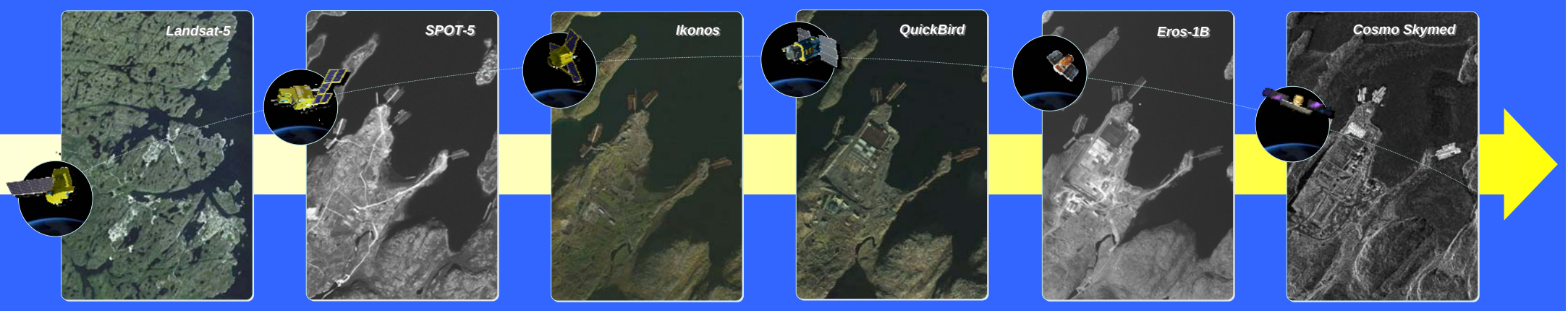
EUSC PRODUCTION



NON PROLIFERATION SECTION

The Non Proliferation Section (NPS) is one of the functional teams within the Operations Division. It´s role is the analysis of installations that are involved, or could be involved, in the spread of technologies used in preparing, or acquiring the capability to obtain Weapons of Mass Destruction (Nuclear, Biological, Chemical). High resolution optical, near infra-red and Synthetic Aperture Radar (SAR) satellite imagery are the main resources used to analyse and monitor nuclear facilities. The combination of remote sensing, imagery intelligence, open source information and their integration into Geospatial Information Systems (GIS) are all techniques used by the experienced Imagery Analysts of this section to deliver comprehensive analyses to EUSC Users.

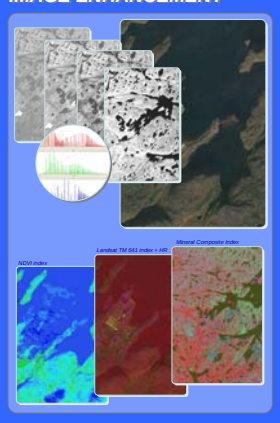
SITE MONITORING



The analysis of a nuclear related installation requires long term monitoring from the first stage of the construction phase. Underground structures, connections and access routes, nature of the terrain, construction materials employed and the overall structure of the facility are some of the crucial elements to be considered for accurate analysis. The use of a full range of satellites and sensors with different spatial resolutions, wavelengths, incidence angles and acquisition times provides an optimal temporal, spectral and spatial analysis capability.

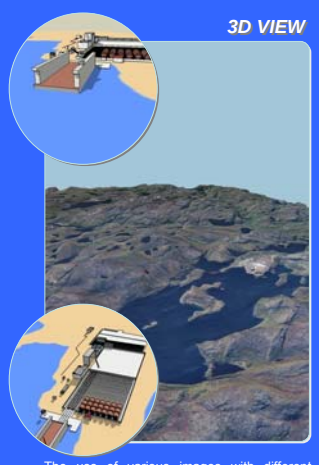
REMOTE SENSING TECHNIQUES & ANALYSIS

IMAGE ENHANCEMENT



Histogram manipulation and image filtering are essential operations at the start of an analysis of any type of satellite imagery. Hyperspectral and multispectral images enable a wider range of processing techniques such as the one illustrated above.

3D VIEW

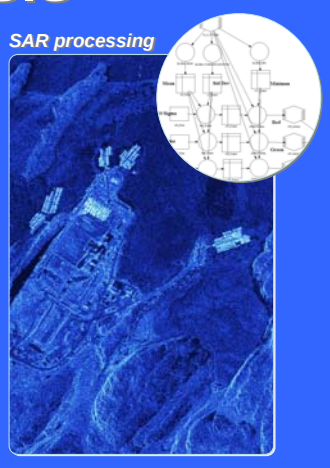


The use of various images with different acquisition angles enables the generation of Digital Elevation Models (DEM) with a high degree of precision. Depending of the spatial resolution of the images, 3D models of the facility structures can also be extracted.

VECTOR - GIS

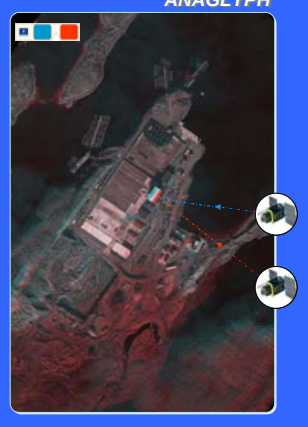
The extraction of vector layers over multi-temporal images and their integration into Geospatial Information Systems (GIS) together with other complementary information is a fundamental aspect of monitoring analysis.

SAR processing



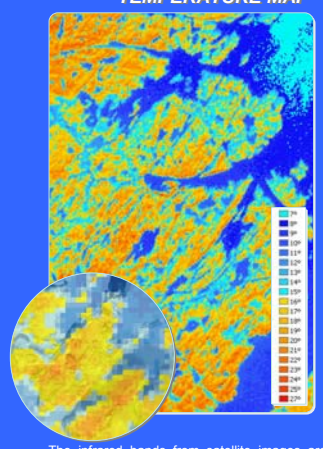
Synthetic Aperture Radar (SAR) satellite imagery with high spatial resolution and multi-polarisation capability has recently become available. Processing this complex imagery using specialist software enables the analyst to extract information that is complementary to that obtained from optical sensors.

ANAGLYPH



Within the range of simple visualization techniques to extract valuable information from satellite imagery, the generation of anaglyph images is one of the simplest and fastest way of producing a 3D visualization, enabling the analyst to gain a better perception of the monitored site.

TEMPERATURE MAP



The infrared bands from satellite images are processed to derive calibrated spectral radiance data. These values are used to generate a surface temperature map to produce an additional information layer to complement the other forms of geospatial data.

A wide range of remote sensing techniques can be applied in order to monitor nuclear related sites. Panchromatic high resolution images are extensively used for detailed analysis and can be combined with multispectral bands to produce high resolution natural or pseudo colour images. The new generation of Synthetic Aperture Radar imagery provides increased opportunities to re-visit monitored sites in all weather conditions and is available day or night. Vegetation and mineral indexes can also be computed from multispectral images in order to obtain soil maps. Temperature maps are processed from Landsat 7 (Band 61/62) or the Aster infrared bands. Satellite images from different acquisition angles enable the production of anaglyph images and the computation of Digital Elevation Models or 3D objects for integration into the GIS products that are now considered essential tools for the visualization and further analysis of geospatial data.

COLLATERAL INFORMATION - SUPPORT

Plans & drawings

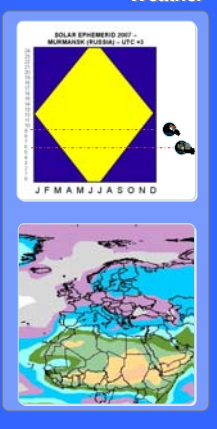
Reports / Documents

Maps

Ground Photos

Technical Documents

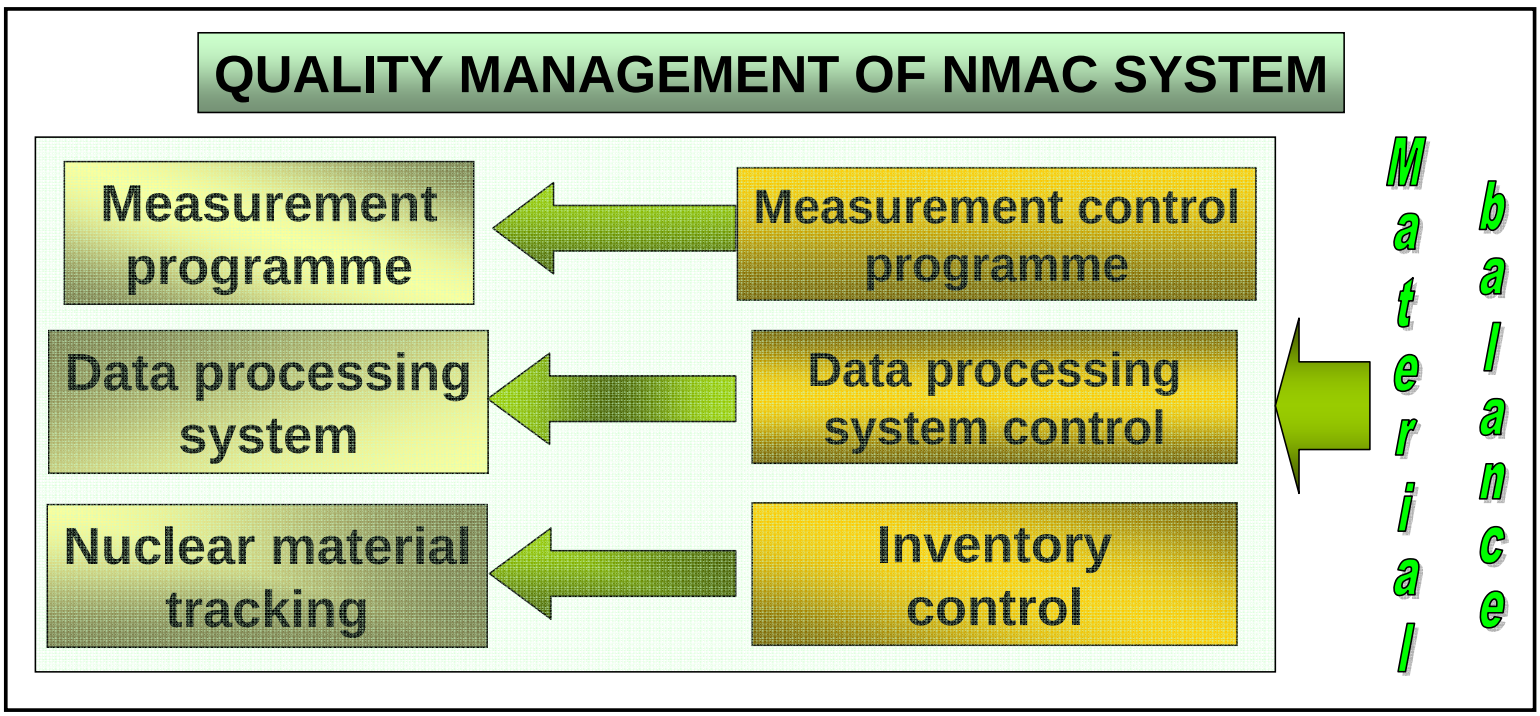
Weather



The analysis of satellite imagery is supported by a wide range of collateral data including open source information. Maps and charts, where available, are particularly useful, as are descriptive plans or drawings. Technical books and other documents serve not only as important aids to imagery analysis but also as reference material for the IAs to maintain and increase their understanding of the complex technologies that lie behind what they can see. Ground photography, if available, can often confirm the hypotheses derived from imagery analysis. Weather conditions over a site and other open sources of information can also help refine the imagery analysis.

THE CONTRIBUTION OF ESARDA TO THE IMPLEMENTATION OF NMAC SYSTEMS AUDITS BY THE EUROPEAN COMMISSION

NMAC SYSTEM MODEL



ESARDA CONTRIBUTION

ESARDA NMAC AUDIT FOCUS WORKING GROUP
Chair: B. Burrows

Guidelines for good Practice in NMAC systems

Advisory report

Guidelines for NMAC audits

ESARDA NMAC AUDIT WORKING GROUP
Chair: R. Weh

Audit criteria interpretation

Information Exchange. Operators and national authorities expertise

Cooperation. Constructive advice to the European Commission

NMAC SYSTEMS AUDIT IMPLEMENTATION

COMMISSION RECOMMENDATION
On the implementation of an NMAC system by nuclear operators

European Commission

LB-NA-24038-EN-N
DOI: 10.2760/19232