INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR

# ITER NEWSLETTER

VOL.3, No.5 MAY 1990

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, AUSTRIA

INSIDE



## HIGHLIGHTS OF ITER COUNCIL MEETING

**by P.N. Haubenreich, ITER Council Secretary**

With a view toward the future

The sixth meeting of the ITER Council included, as usual, review of technical progress and plans for completion of Conceptual Design Activities. However, at this meeting the principal direction of thought was toward the future. At its previous meeting, the Council had concluded that the appropriate next technical step should be engineering design and that it would be highly desirable to avoid any unnecessary hiatus after completion of the CDA. Toward this aim, the Council suggested in December 1989 that "the Parties consider entering discussions with a view toward negotiations of an instrument to allow Engineering Design Activities." At its sixth meeting, on 26-28 April in Vienna, the Council was pleased to hear from all delegations expressions of willingness to enter the suggested exploratory discussions. Considering the schedules of internal reviews taking place in several of the Parties' home organizations, it was the Council's understanding that these discussions could begin as early as the end of July 1990.

The Council drafted a Status Report, summarizing pertinent information as of the end of April. It also took steps to generate more information that should be useful in the discussions. In particular, it initiated additional studies by its Ways and Means Working Party to further clarify several important issues and options. It is expected that results of these studies will be available before the end of July. Meanwhile the ITER Management Committee will have completed preliminary estimates of costs and schedule of possible Engineering Design Activities (EDA).

Canadian interest in hosting EDA

During the meeting, the Council received communications from the Governments of Canada and Ontario, and Ontario Hydro, reporting a study undertaken at their initiative, which favourably reviews the feasibility of the Province of Ontario hosting ITER Engineering Design Activies if the Parties should agree to

Discussion at the ITER Council Meeting, Vienna, 26 - 28 April 1990 proceed. Several of the Council delegations noted that institutions within their countries were interested in the possibility of hosting the EDA work site. The Council expressed its appreciation for the interest and support from Canada and, while noting the broad interest, stated that any questions of site would be a matter for the Parties in their consideration of possible EDA.

## ITER SPECIALISTS' MEETING ON MAGNET MATERIALS

**by John R. Miller, Head, Magnet Design Unit**

At the beginning of the ITER Conceptual Design Activities (CDA), a data base was constructed to be representative of the performance attainable from critical magnet materials: primarily superconductors, insulators, and structural alloys. Since significant progress has been made, both in the design and analysis of the ITER magnet systems and in the materials R&D, a meeting of materials specialists was convened in Garching in February 1990 to advise the Magnet Design Unit on an appropriate reassessment of that data base against the present ITER design assumptions and requirements. The meeting had a second, equally important objective, which was to discuss the needs and the appropriate organization for materials R&D during possible ITER Engineering Design Activities (EDA).

**Structural alloys**

For steels, there are two distinct categories of requirements:

1. thick, rolled plates or forgings for the Toroidal Field (TF) coil cases, and
2. conduits for force-cooled conductors, which may be formed from thin sheet.

From discussions at the meeting, it was clear that several choices in each category appear consistent with present assumptions and requirements, but the needed characterization of these materials is far from complete. For case alloys, it must be demonstrated that reported properties are consistent with the methods appropriate for producing them in large heats and thick sections (nearly 400 mm thick required) and with welding these sections together to form a 400 ton TF coil case. For the conduit steels, the levels of strength, toughness, and fatigue resistance assumed appear to be attainable in several candidate materials, but all the materials are more or less sensitive to important processing variables, e.g. to the requirement in many conductor designs that the conduit be subjected to the same heat treatment as the $Nb_3Sn$ conductor strands.

For both categories of materials, characterization near 4 K is essential and the sensitivity of properties to details of the alloy composition and other process variables must be well established during the EDA. Also, to establish the fatigue life of critical components, the appropriate non-destructive examination and repair techniques for flaws in finished parts must be developed.

**Superconductors**

Both NbTi and $Nb_3Sn$ may have application in ITER magnet systems, but $Nb_3Sn$ has a clear advantage when force-cooled conductor options are selected because of the higher temperature margins provided. In the present designs for the central solenoid and the TF coils, $Nb_3Sn$ is the only sensible choice.

The performance assumed during the CDA for $Nb_3Sn$ strands includes a detailed accounting for its multiparameter dependences, but it can be summarized, though somewhat simplisticly, as follows: critical current density of 800 $A/mm^2$ over the non-stabilizer cross-section of strands at 12 T, 4.2 K, with no externally applied strain, and no damage; ac hysteresis losses consistent with effective filament diameters around 10 $\mu$m; and nearly undiminished critical current density after $10^{23}$ neutrons/$m^2$ fluence.

Bronze-process wires offer good ac-loss performance but generally fall somewhat short of the critical current goal. Tube-process wires meet the goal but have too large filaments. Internal-tin-process wires have, in some instances, met all the assumed performance goals. Concerns about the latter include:

1. sometimes when process variables are adjusted for high current capacity, the filaments become effectively bridged and losses are drastically increased, and

2. the technology is relatively immature in terms of total production so that quality and uniformity of the product in large quantities need to be proved.

Production of some tens of tons may be needed during the EDA to prove the process, but such quantities are quite in line with developmental production for other large projects.

**Insulations**

We expect insulations in ITER magnet designs to permit operation with 20 kV terminal-to-terminal and terminal-to-ground. In addition, winding-pack insulations are expected to tolerate, with adequate margins, up to 450 MPa compressive stress normal to laminations, +/- 0.25% strain parallel to laminations, and 30 MPa shear stress in the plane of laminations. In the TF coils, these allowables must be retained after exposure to 50 MGy locally and 20 MGy over broader regions of the windings. There do appear to be candidate insulation systems consistent with these allowables. Although some of the promising candidate materials have been tested only after irradiation to 5 MGy, there is no degradation apparent at that level, and a programme is in place to continue testing after irradiations up to 50 MGy. It is essential that validation during the EDA of the chosen insulation system result from irradiation at essentially 4 K with the appropriate mix of neutrons and gammas, followed by mechanical testing at 4 K. Important facilities having the appropriate capabilities were identified during the workshop.

---

Editor's Note

*Encouraging progress of the ITER Conceptual Design Activities focuses more and more the attention of fusion scientists and engineers of many countries. It is understood that many of them, having contributed much to the fusion development in the past and continuing to carry out various important research at present, are willing to share their thoughts on this joint international venture and on how the collaboration might be further promoted. An original view point on the prospectives of fusion research and the Next Step devices being now in the conceptual design phase, including ITER, is given in the following article written by a representative of a developing country, an eminent Indian fusion scientist, Prof. P.K. Kaw.*

---

## ON THE PACE OF FUSION RESEARCH - A VIEW POINT

**by P.K. Kaw*, Director, Institute for Plasma Research, Bhat, India**

**Rapid development in the past decade**

Fusion research has made major technical strides in the past decade or so. With the impressive experimental demonstration of magnetically confined plasmas with an equivalent $Q_{DT}$ of 0.8, scientists are today at the threshold of "scientific break-even" experiments - a long time dream of fusion researchers. The ultimate target of the research is, of course, the construction of prototype reactors with a view to their industrial production, marketing, and wide development. To realize this target, one must yet cross the following milestones:

**Milestones yet to be achieved**

1. Production and control of long time (preferably steady-state) <u>ignited</u> plasmas with proper design and utilization of plasma facing components (first wall, divertor, etc.) and superconducting magnets.

2. Demonstration of major blanket functions like tritium breeding, high-grade heat removal, shielding, etc.

3. Optimization of fusion reactor economics involving development of materials having proper life-time end safety-environmental aspects, component life-time evaluation, maintenance and decommissioning procedures, efficient electricity production, etc.

* Prof. P.K. Kaw is a member of the International Fusion Research Council (IFRC)

Role of Next Step devices

The Next Step devices, which are presently in the conceptual design phase, would address major questions, related to the first two milestones. In this context, it is noteworthy that much of the blanket technology mentioned under item 2 above could be directly borrowed from extensive developments in fission reactor systems. As the Next Step device achieves optimal performance, the stage would be set for the construction of a demonstration protoreactor which would address the residual issues discussed under the third milestone above.

At this point it is pertinent to ask the question "Now that scientific breakeven experiments are in the offing, how rapidly must the subsequent research and development for fusion reactors be carried out?" The desirable pace of further R&D in fusion is very much a function of the perceived need for fusion energy and the economics of the final product vis-a-vis other energy production systems. The perceived need will in turn depend upon our assessment of the energy scene in the world. If one <u>only</u> looks at the energy scene in the developed world, one is likely to get a feeling of complacency and regard development of fusion energy as a luxury which can wait for another 50 to 60 years. However, such a view would be highly short-sighted for the following reasons:

Taking into account energy situation in the developing world

1. There is a yawning gap between energy production and energy needs in the developing world (Asia, Latin America and Africa). Taking India as an example, it will be generating about 100 GW of electricity by the year 2000, giving it a *per capita* consumption less than 10% of that of a developed European nation like France. Granted that new future ideas about energy conservation might reduce the *per capita* energy requirement to some degree - the question still remains: how will India and other developing nations fill the growing energy demands?

2. If this energy gap in the developing world is filled by burning fossil fuels, the consequences to the global environment in terms of greenhouse effect, acid rain, etc. are staggering!

3. How realistic is it to assume (as is often done) that other benign energy production methods like hydroelectric, solar, renewable energy sources, etc. can fill the gap in the developing world? India's assessment is that it is too naive a view of realities. Thus there is an enourmous energy market in the developing world which fusion reactors could readily and profitably tap provided they can be made economically competitive. This assessment of economic competitiveness must make a careful analysis of the environmental aspects and clean-up costs of all the viable options.

4. The present comfortable energy position in the developed world is itself a function of the political situation in the oil producing countries (remember the shock of the 70's).

To keep momentum in fusion research

5. Finally fusion research has gained a momentum which it will be ill-advised to slow down. If we portray fusion reactor development as a 50- to 60-year-enterprise, it will not attract any bright people in the future; the danger is that existing highly motivated teams will disperse.

From the above discussion it is clear that fusion energy reactors should be developed at as rapid a pace as is technically feasible. Furthermore, optimization of reactor economics and reduction of electricity generation costs should be an integral part of this development.

We now concentrate our further discussion on the immediate Next Step device, as examplified by ITER, which is presently undergoing conceptual design and technically could be ready for experimentation early in the decade beginning with the year 2000. As already mentioned, such a test reactor would achieve ignition and control of a burning D-T plasma for long-time periods and would also investigate a number of technological issues related to superconducting magnets, first wall and divertor development, blanket shielding and tritium breeding, etc. It also seems appropriate that this test reactor be built via substantial international co-operation among the world's largest fusion programmes. However, the following point must be emphasized. To optimize the performance of the international test reactor, a considerable amount of parallel research and development will have to be carried out by strong national programmes all around the globe. Thus a great deal of interaction and co-operation has to be fostered among all the significant programmes in the world and this is a task which the IAEA must take up in earnest. IAEA can foster wider international participation in ITER by encouraging smaller programmes to submit proposals on ITER R&D tasks to the appropriate ITER entity and letting it make a selection out of them after careful technical scrutiny.

World-wide support of ITER R&D is advised

We conclude that the overall pace of fusion research should be accelerated rather than slackened because:

- The programme is making major technical strides and has gained a momentum which it will be ill-advised to lose.
- A good part of the world will greatly benefit by an early deployment of fusion energy.
- Investments in this area will more than pay for themselves because of the huge energy market which can be tapped, cleaner global environment that will undoubtedly result and the inherent safety of such systems.

Investments will more than pay themselves

---

Editor's Note

*The Newsletter presents here below information on the activities of two more national research centres contributing to ITER design. These are Japan Atomic Energy Research Institute (JAERI) and the Oak Ridge National Laboratory of the United States.*

---

## FUSION RESEARCH AT JAERI

**by M. Tanaka, Naka Fusion Research Establishment, JAERI**

Broad fusion research is ongoing at JAERI

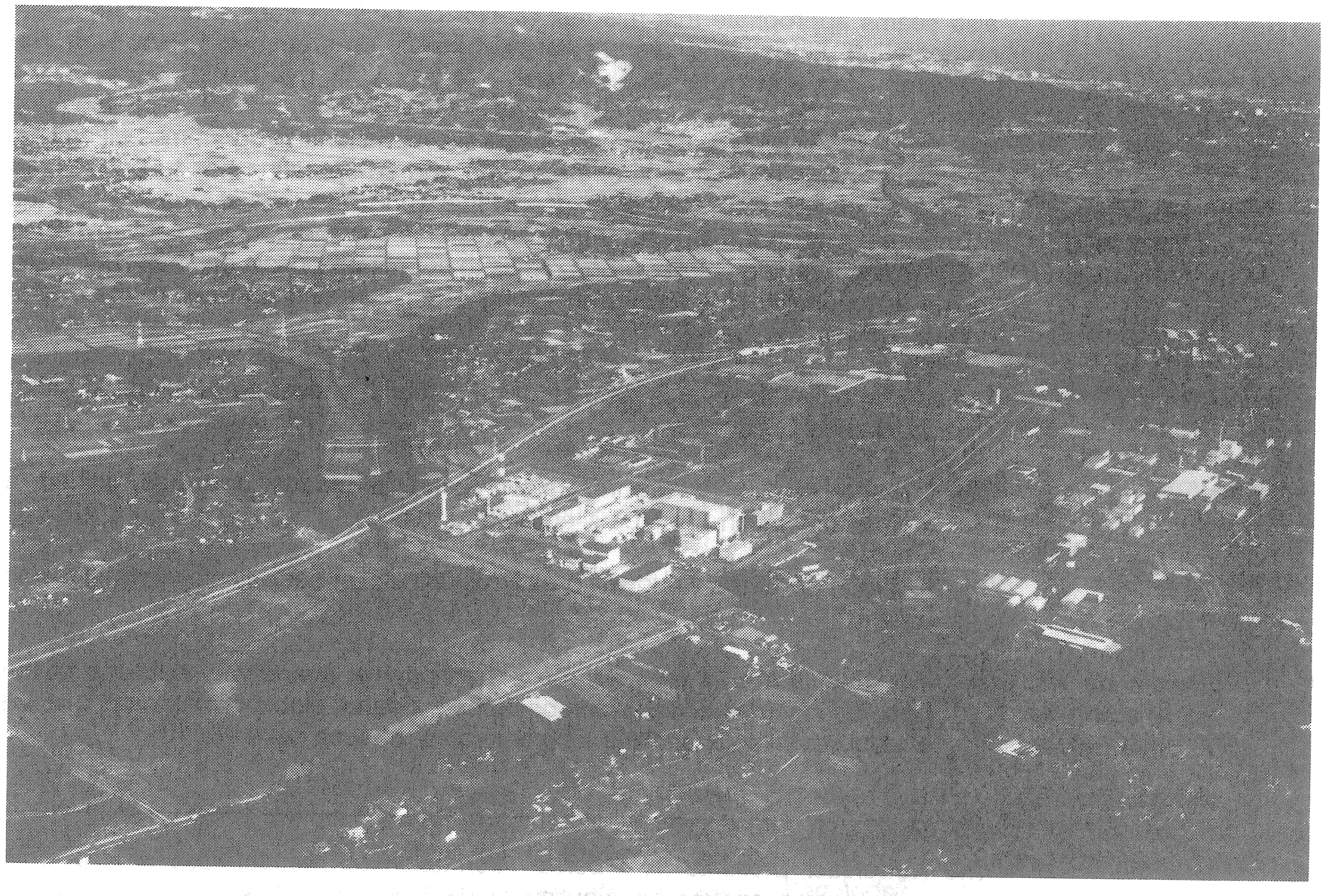
A broad research programme on fusion is ongoing at JAERI (Japan Atomic Energy Research Institute), especially at Naka Fusion Research Establishment, which is the dedicated research centre for fusion of JAERI. The other research centres, including Tokai and, also, Oarai and Takasaki, direct their research efforts to fusion on the basis of their long experience in fission technology.

National activities of Japan on ITER are the responsibility of the Fusion Experimental Reactor (FER) Team at the Naka Fusion Research Establishment. The role of JAERI and the respective organizational structure were described in the ITER Newsletter, Vol. 2, No. 1. The purpose of this article, therefore, is to present some basic fusion research activities of JAERI, including recent developments, which may have impact on the ITER design.

JAERI tokamaks

JT-60 and JFT-2M are the two well-known tokamaks located at Naka. After completion of the scheduled experimental programme at the end of October 1989, JT-60 is now under large-scale remodelling. The vacuum vessel and all poloidal coils are being replaced by newly manufactured ones. The maximum plasma current expected in the new tokamak, JT-6OU, is 7 MA (limiter) and plasma elongation will be up to 1.8 with lower single-null divertor configuration. The JT-6OU experiment will begin early in 1991 with an increased NBI power of 40 MW. In the course of the experiment, installation of parallel NBI lines, new launchers for LH and IC, and new diagnostic instruments is scheduled.

Experiments on two regimes of energy confinement, H- and improved L-modes, are being continued on JFT-2M. Magnetic ergodic limiters have been applied recently



Naka Fusion Research Establishment

and now a limiter bias experiment is in preparation. Also, a FW current drive experiment is being made in combination with ECH. To increase the toroidal field, which is of maximum 1.5 T at present, a shared use of one of the JT-60 power supplies is planned.

**Plasma heating**

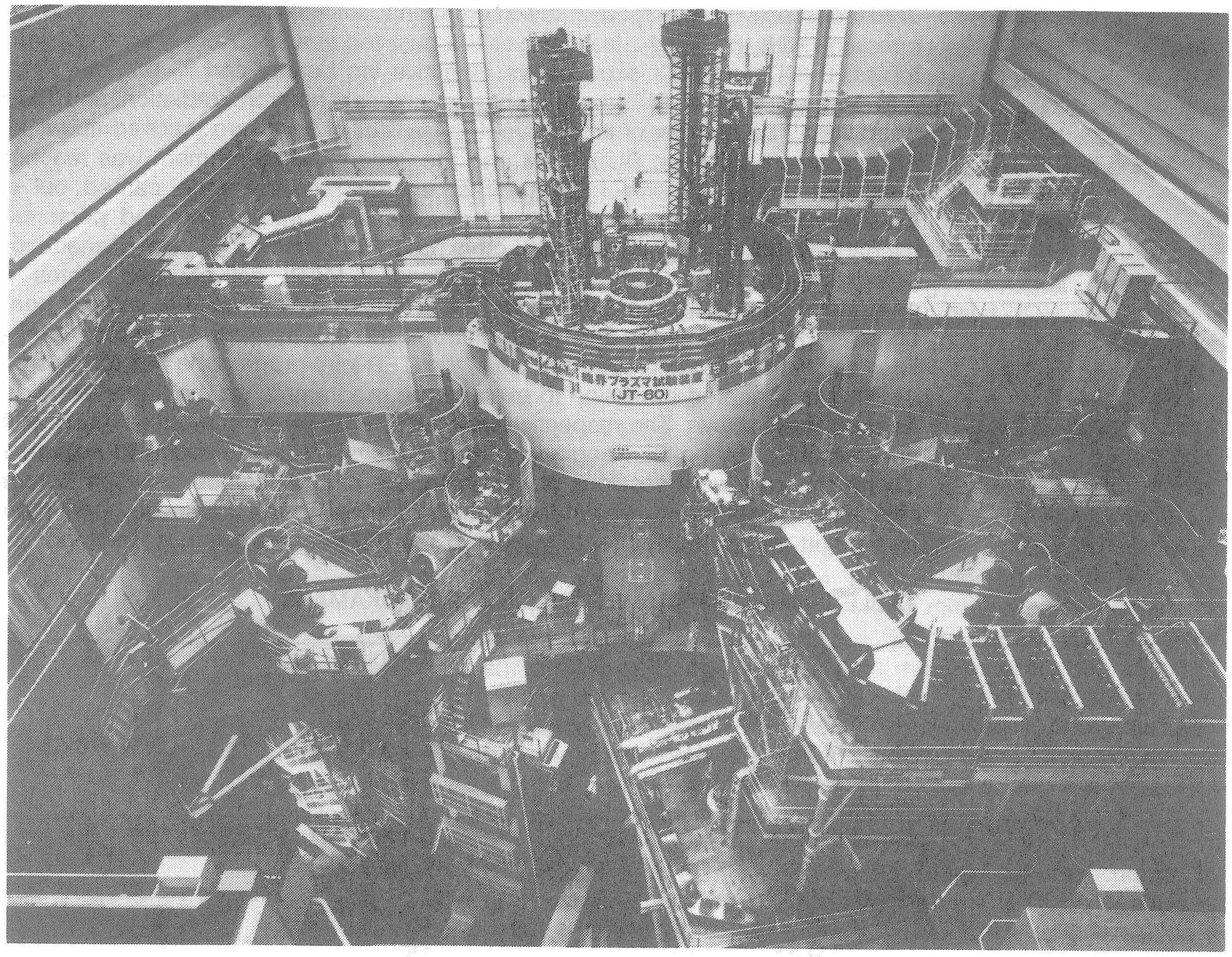
Development of high-current negative ion beams has been advanced. 7.8-A beams were extracted at 75 keV, limited only by the capacity of the test stand. In addition, a basic acceleration experiment is being made at 350 keV but with low currents, about 0.1 A. A test-fabricated 120-GHz whispering-gallery-mode gyrotron revealed short pulse (1 ms) performance with an output power of 520 kW. In a preliminary experiment, a 3-kA, 1-MeV, and 100-ns electron beam was obtained using an induction linac core assembly, which was recently completed.

**Plasma facing components and vacuum**

A 4-A, 100-kV electron beam irradiation test facility, JEBES, was completed showing a flexible operation performance, from 1 ms pulses to continuous beams. For Model Divertor plates, various carbon materials including carbon-fibre-composites both with and without metal heat sink, were tested, as well as refractory metals with Cu heat sink. Ion beams from JT-60 NBI prototype were also used for the purpose. A test-fabricated turbomolecular pump with ceramic rotating blades ($Si_3N_4$, 21 cm diameter, and 25,000 rpm), driven pneumatically and gas-sealed, demonstrated expected performance in combination with a ceramic fore-pump. A scale-up to larger diameter blades (30 cm) is expected late this year.

**Superconducting magnets**

Experiments are ongoing in the Superconducting Magnet Laboratory at Naka on model centre solenoid coils with an inner bore of 1 m. Pulsed operation utilizes the JT-60 ohmic power supply ($\pm$5 kV, 58 kA). A $Nb_3Sn$ pancake coil reveals excellent performance, while a pair of NbTi coils need further amelioration. Experiment on a US-made pancake ($Nb_3Sn$) is scheduled.




Tokamak JT-60

Test fabrication of several types of high-current (30-kA) conductors for toroidal coils of next-step tokamak was completed. Winding a pancake will be made shortly.

**Fuel cycle and blanket**

Basic processes related to fusion fuel cycle, such as removal of chemical impurities from reactor exhausts, separation of tritium from hydrogen isotope mixtures etc., have been studied in the Tritium Process Laboratory facility since 1988. Also important in JAERI's programme for fusion fuel cycle development is the participation in operation of the Tritium Systems Test Assembly (TSTA) at Los Alamos, USA. A new fuel clean-up system provided by JAERI is now under installation on TSTA loops. The test-production of tritium at a level of 100 Ci has been made by irradiating LiAl alloys using Japan Materials Testing Reactor at Oarai.

Experiments on neutronics using model blanket assemblies are being made on the Fusion Neutronics Source (FNS) facility, which features a $5 \times 10^{12}$ neutrons/sec (RTNS-type rotating target). A part of the experiment is made as joint programme with the U.S.A.

**Structural and tritium breeding materials**

The heavy irradiation test of candidate alloys for fusion had been made on fission reactors, in particular on HFIR and ORR in Oak Ridge, U.S.A. Broad basic studies, on low-induced-activity alloys, etc. are continuing. As for tritium breeding materials, emphasis has been placed on basic studies of Li ceramics, oxide, aluminate and silicate. The study covers in-situ tritium release measurements on JRR-2 in Tokai. In addition, as an international collaboration, a scale-up experiment is scheduled on the FFTF facility at Hanford, U.S.A.

International collaboration

JAERI has actively engaged in the international collaboration on fusion. The co-operation with the U.S.A., lasting over the past ten years, covers a wide range of topics: D-III-D and MTX experiments, negative ion beams and IC tetrodes tests, pulse coil experiment, FNS experiment and materials neutron irradiation, and TSTA operation. The number of items involving transfer of hardware reached about 10. An agreement between EURATOM and the Japanese Government was reached early last year. Experiments on direct energy recovery were carried out at Cadarache, France, using JAERI's plasma source. Information and personnel exchanges have been made with other countries, especially with Canada on tokamak experiment and tritium technology. JAERI actively participates in multilateral co-operation organized by IEA. The Large Coil Task has already been completed; and Breeder Exchange Matrix II (Beatrix II) started recently. In the international collaboration through IAEA, the ITER joint programme has top priority. JAERI has accepted a mission to execute the programme on the Japanese side, as mentioned above The respective task force, FER Team, comprises JAERI staff supplemented by on-loan engineers from industry. It is interesting to note that, during the joint work sessions at Garching, the time difference between Europe and Japan, eight hours, was indeed beneficial allowing the Japanese home team to support expeditiously the daily work of the ITER central team at Garching.

TABLE I. MAJOR PARAMETERS OF JAERI TOKAMAKS

| | JT-60 | JT-60II * | JFT-2M |
|---|---|---|---|
| Major radius (m) | 3.0 | 3.4 | 1.3 |
| Minor radius (m) | 0.95 | 1.1 x 1.5 | 0.35 x 0.53 |
| Toroidal field (T) | 4.5 | 4.2 | 1.5 |
| Plasma current (MA) | 2.7 (divertor) | 6 (divertor) | 0.4 |
| | 3.5 (limiter) | 7 (limiter) | |
| Heating (MW) | | | |
| NBI | 26 | 40 | 2 |
| LH ** | 10 | 15 | |
| IC ** | 3 | 10 | 2.5 |
| EC ** | | | 0.4 |

* will be operating in early 1991
** output power of generators

## THE OAK RIDGE NATIONAL LABORATORY (ORNL) *

by C.C. Baker, E.E. Bloom, M.J. Gouge, J.T. Hogan, R.C. Isler, D.C. Lousteau, N.A. Uckan, D.W. Swain

ORNL missions and resources

ORNL stands today as the United States' largest and most diverse energy research and development (R&D) institution. The Laboratory has two primary missions. One is to conduct applied research and engineering development in support of the U.S. Department of Energy (DOE) programmes in energy conservation, fusion, fission, fossil, and other energy technologies. The other primary mission is to perform basic scientific research in selected areas of the physical and life sciences.

A secondary mission is to apply the Laboratory's resources to other nationally important tasks such as international competitiveness, hazardous wastes, and selected areas of national defense. In addition to the R&D roles, ORNL designs, builds, and operates user facilities for the benefit of university and industrial researchers and supplies radioactive and stable isotopes that are not available from industry.

* Operated by Martin Marietta Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

The Laboratory's programmes, which total over $ 400 million per year, are funded by a number of sponsors. DOE, the owner of the Laboratory, is ORNL's largest sponsor, providing nearly 80% of total funding. The number of staff is about 4400. In fusion, the emphasis is on stellarator confinement configurations, plasma heating, fuelling systems, first-wall and blanket materials, applied plasma physics, and fusion engineering design activities. The latter provides major support to the Compact Ignition Tokamak (CIT) and ITER.

**ORNL role in ITER**

ORNL is involved in a broad range of ITER design activities, including overall tokamak configuration design, maintenance/assembly, in-vessel components (divertor, first wall, blanket, shield), vacuum vessel, cryostat, ICH systems, and pellet fuelling systems. These are described further in this article, as well as ORNL's contribution to physics studies for ITER. Validating R&D support of ITER includes materials work on austenitic steels and graphite/carbon composites and advanced pellet injectors. Most of the ORNL fusion programme supports ITER either directly or indirectly.

**Engineering design and analysis**

The Fusion Engineering Design Center (FEDC) at ORNL is applying an integrated approach to the design of the ITER blanket/shield modules. Using the combined expertise of resources available in Oak Ridge, the key design issues are being simultaneously addressed. A significant effort is being directed toward accurately characterizing the loads induced in the modules during plasma disruption and incorporating them into the structural design of the modules and their mounting system. Assembly and maintenance studies of the effect of module geometry and associated segmentation scheme on the assembly sequence and equipment requirements are being carried out in parallel with this effort. The FEDC also supplies the mechanical design support for the U.S. Nuclear Group working on the blanket and shield concept to ensure compatibility with the overall concept.

The Tokamak Simulation Code (TSC) is being used to model the interaction of the plasma motion and current decay during disruption events with the conducting surfaces in a self-consistent manner. The code identifies the time evolution of plasma parameters and poloidal fluxes. Using these results, a code developed in Japan (EDDYCUFF) is used to calculate the resulting eddy currents and forces in reactor components. The output of this code is then input into a finite element stress analysis code to analyze the resulting stresses and identify loads at required interfaces. The TSC code may also be used to study the sensitivity of the forces to the current decay time and to initial conditions to determine the disruption scenario that produces the most severe loads. Fig. 1 shows typical eddy current patterns in the inboard and outboard blanket/shield modules during a plasma disruption.

Engineers experienced in design of remote handling systems for nuclear applications at ORNL are developing concepts for handling of the blanket modules for assembly and maintenance. The study is addressing not only the procedures and equipment required to accurately position the modules within the vacuum vessel, but also methods to install the hardware that fastens them to the vessel and connects the coolant lines. All efforts are aimed at presenting an integrated design of the blankets, their installation scheme, and an attachment method consistent with an overall in-vessel configuration developed by the ITER designers.

**Materials**

The Fusion Materials R&D programme at ORNL is focused on reactor-structural alloys, ceramics, copper alloys, and graphite and carbon/carbon composites for first wall and blanket structures. A common theme that ties all of our materials research together is the investigation of the effects of neutron irradiation on the properties of these materials.

Present research on structural alloys includes austenitic stainless steels, ferritic or martensitic steels, and vanadium alloys. Of particular importance to the ITER project is the research on austenitic stainless steels which is conducted in collaboration with the Japan Atomic Energy Research Institute. Confident prediction of the mechanical performance and dimensional stability of structural

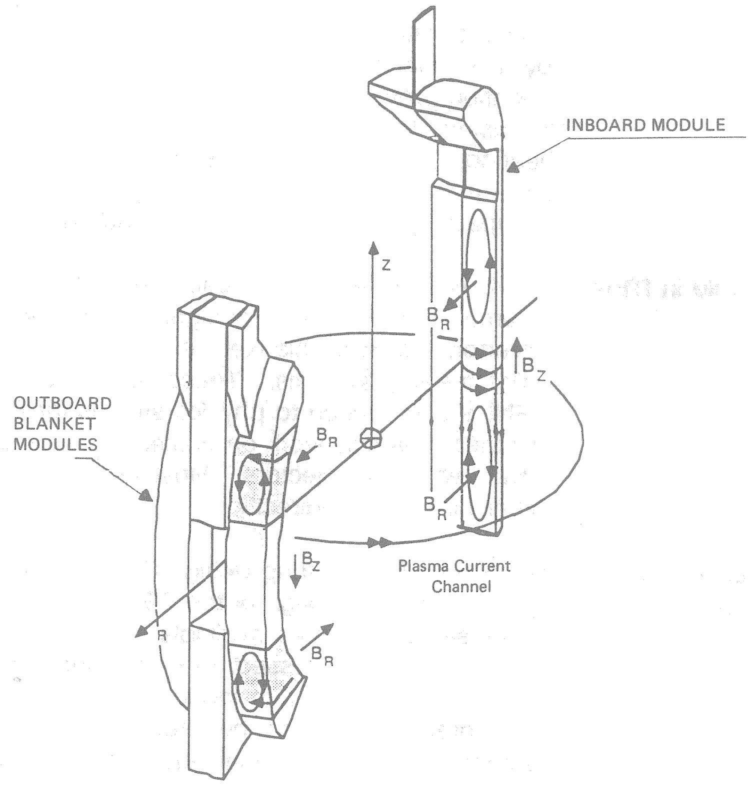



Fig. 1. Eddy current patterns developed in blanket modules during plasma disruption.

alloys requires irradiation experience at temperatures, stresses, damage rates, and fluences characteristic of the application. In the absence of either a 14-MeV neutron source or a fusion reactor, the closest approximation to the fusion, and specifically the ITER, environment can be obtained through spectral and isotopic tailoring experiments, both of which are presently being pursued. The spectral tailoring experiment makes use of the two-step reaction with thermal neutrons, $^{58}Ni(n,\gamma)^{59}Ni$ and $^{59}Ni(n,\alpha)^{56}Fe$, to achieve He/dpa ratios similar to that produced in a fusion spectrum.

Irradiations are conducted in the High Flux Isotopes Reactor (HFIR), where the thermal neutron flux is periodically reduced to give approximately the helium generation rate in the fusion spectrum. Since atomic displacements are created by the fast neutrons, the dpa production rate remains essentially unchanged during the experiment and, for these experiments, it is approximately equivalent to a neutron wall loading of 1 MW/m$^2$ or very close to the ITER conditions. Controlled irradiation temperatures of 60, 200, 330, and 400°C are achieved.

For the austenitic stainless steels, properties being investigated include swelling, irradiation creep, tensile fatigue, and irradiation assisted stress corrosion cracking. Several alloy compositions are included in the experiment. Some of the most interesting and important results are in the area of irradiation creep. Creep rates are found to be dependent on the He/dpa ratio. Irradiation creep is a significant deformation mechanism at stresses and temperatures relevant to ITER (Fig. 2). Isotopic tailoring, in which the ratio of $^{58}Ni$, $^{59}Ni$, and $^{60}Ni$ are varied in the initial alloy so as to achieve the desired He/dpa ratio, is being used to investigate irradiation effects at damage levels approaching 100 dpa. Isotopic tailoring experiments are conducted in the HFIR target positions in which the fast and thermal fluxes are much higher than in the positions in which the spectral tailoring experiments are conducted.

Graphite and carbon/carbon composites will be used extensively in the first wall and divertors of fusion reactors where low-Z and thermal shock resistance are essential. A major limitation with these materials is the deleterious effect of neutron irradiation on critical physical and mechanical properties. A programme is in progress to determine not only the magnitude of these effects but eventually to design materials with properties optimized for the fusion environment. Low fluence irradiations will be conducted in the test reactor at the Buffalo University to investigate changes in thermal conductivity and electrical resistivity. The effects of irradiation on dimensional stability and mechanical properties are being investigated with irradiations in the FFTF and the HFIR.

There is currently no information on the effects of irradiation in the fusion neutron spectrum and the resulting irradiation damage on the properties of ceramics. To achieve the He/dpa ratio appropriate to the fusion spectrum, we are using isotopic tailoring in which the ratio of $^{17}O$ and $^{18}O$ is varied in ceramics such as $Al_2O_3$ and $MgAl_2O_4$, and the materials are irradiated in a mixed spectrum thus generating helium from the reaction $^{17}O(n, \alpha)^{14}C$ and displacement damage from the fast neutrons. In these experiments, which are in collaboration with the Los Alamos National Laboratory, the post-irradiation mechanical and electrical properties will be determined. Other experiments are being planned to investigate in situ properties.

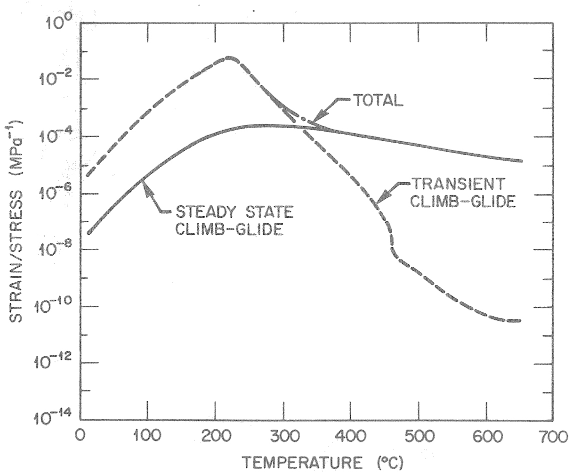



Fig. 2. Recent results from the HFIR spectral tailoring experiments have shown that irradiation creep remains a significant deformation mechanism at temperatures as low as 60°C. This figure shows strain/stress as a function of temperature as derived from the experimental data and fit to models for irradiation creep deformation. Deformation from irradiation creep will be a major consideration in the analysis of the behaviour of the ITER first wall and blanket structure. A major shortcoming is that such information does not exist for alloys other than the stainless steels.

---

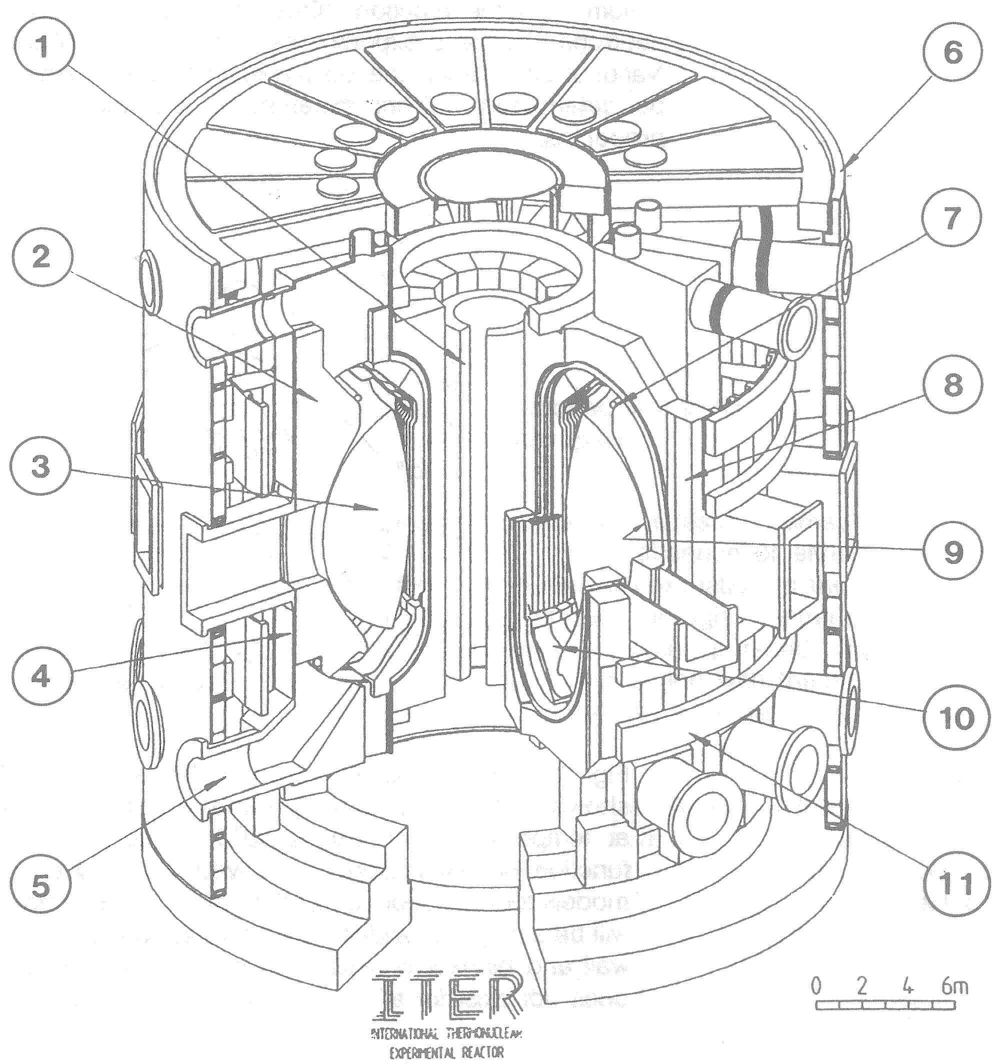
(This article will be continued in the next issue of the Newsletter. The ORNL research on pellet fuelling, FW current drive and plasma theory as well as the experimental studies at the Advanced Toroidal Facility will be described).

## MAJOR ITER EVENTS - 1990

| | | |
|---|---|---|
| Ways and Means Working Party Meeting | Tokyo | 6 - 8 June |
| Joint Work Session | Garching | 2 July - 16 Nov |
| Ways and Means Working Party Meeting | Garching | 10 - 13 July |
| ISTAC Meeting | Vienna | 12 - 14 Sep |
| ITER Council Meeting | Washington | 8 - 9 Oct |
| ISTAC Meeting | Vienna | 28 - 30 Nov |
| ITER Council Meeting | Vienna | 13 - 14 Dec |

## ITER REFERENCE PARAMETERS

| | |
|---|---|
| Plasma major radius, R (m) | 6.0 |
| Plasma half-width at midplane, a (m) | 2.15 |
| Elongation, 95% flux surface | 1.98 |
| Toroidal field on axis, $B_o$ (T) | 4.85 |
| Nominal maximum plasma current, $I_p$ (MA) | 22 |
| Nominal fusion power, $P_f$ (MW) | 1000 |




1- CENTRAL SOLENOID
2- SHIELD/BLANKET
3- PLASMA
4- VACUUM VESSEL-SHIELD
5- PLASMA EXHAUST
6- CRYOSTAT
7- ACTIVE CONTROL COILS
8- TOROIDAL FIELD COILS
9- FIRST WALL
10- DIVERTOR PLATES
11- POLOIDAL FIELD COILS

The ITER NEWSLETTER is prepared and published by the International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria. Telex: 1–12645, Cable: INATOM VIENNA, Facsimile: 43 1 234 564, Tel.: 43 1 2360–6393/6394. Items to be considered for inclusion in the ITER Newsletter should be submitted to N. Pozniakov, ITER Secretariat.

Printed by the IAEA in Austria
May 1990

90-01911