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INSIDE

ITER FUEL CYCLE DESIGN ACTIVITIES

by P.J. Dinner, Fuel Cycle Design Unit Leader

Functions and Interfaces

Mission of Design Unit

The mission of the Fuel Cycle Design Unit is to co-ordinate and integrate the conceptual designs of all processes involved in the extraction, processing and control of tritium and deuterium within the ITER facility. A list of functions is given in Table 1. Interconnections among fuel cycle processes are shown in Fig. 1.

Fuel cycle processes have interfaces to basic machine, breeding blanket, first wall, divertor, and plant auxiliaries such as cooling water, and heating, ventilation and air conditioning. In fact any system which comes in contact with the D-T fuel represents an important design interface. Fuel cycle considerations therefore significantly affect the design of elements located throughout the plant and have an important impact on overall plant design.

Environment and Safety

The total inventory of tritium in the ITER during operation is expected to be somewhat less than 5 kg, of which approximately 1 kg will be involved in processing and the remainder will be either in storage or retained in structural or breeding materials. Careful design of plant systems that contain tritium is essential to meeting the environmental and safety goals of ITER.

TABLE 1. ITER FUEL CYCLE FUNCTIONS

PLASMA FUELLING	BLANKET TRITIUM EXTRACTION
Gas puffing	Driver Blanket
Pellet injection	Blanket Test Modules
PRIMARY VACUUM PUMPING	AUXILIARY PROCESSES
Torus conditioning and pumpdown	Isotopic separations
Plasma exhaust	Fuel storage
	Atmosphere detritiation
FUEL GAS PROCESSING	Waste water processing
Impurity removal	Solid waste handling

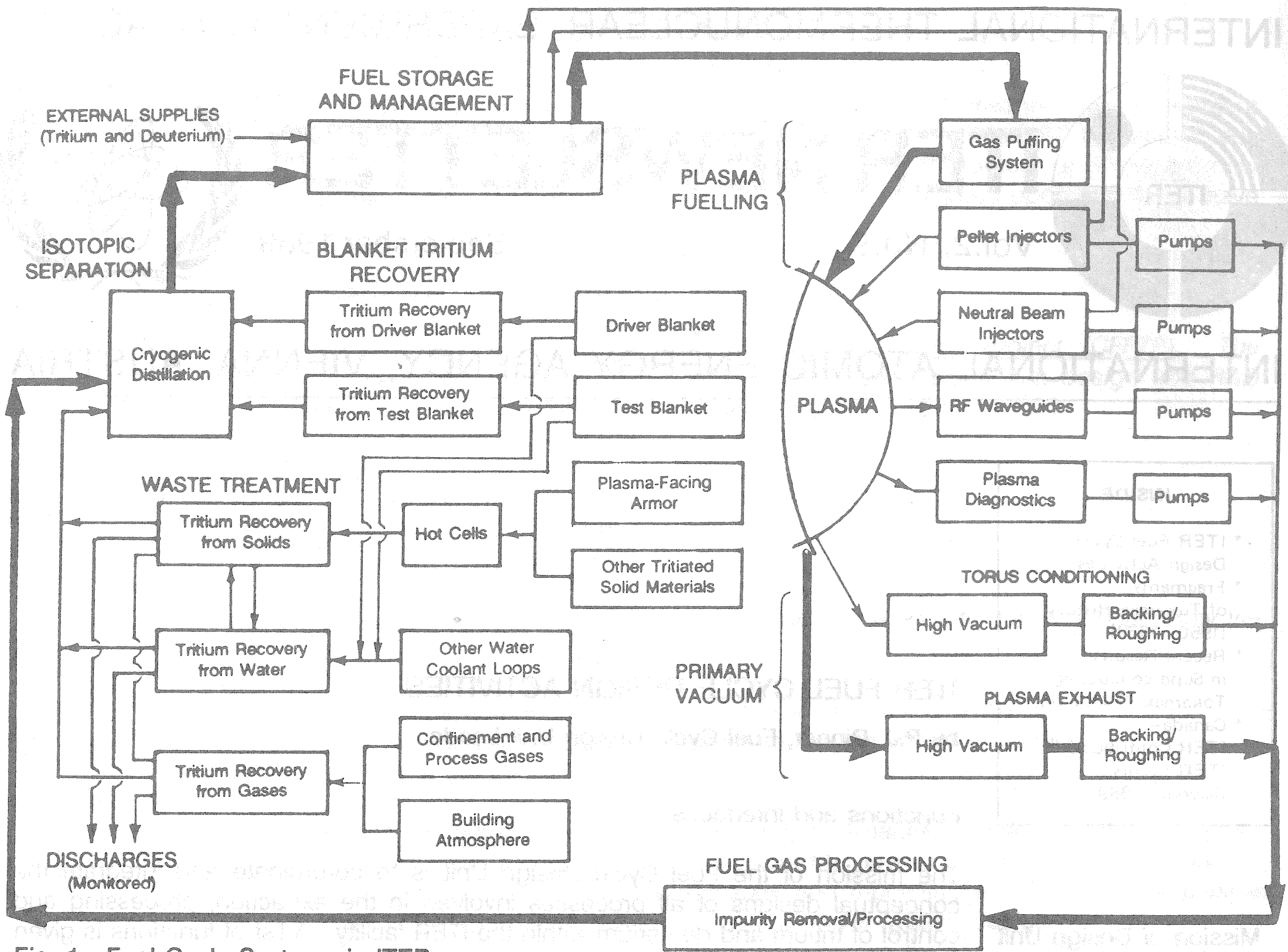

Fig. 1. Fuel Cycle Systems in ITER

Careful design to meet special requirements

Because the fuel is a mixture of isotopes of hydrogen, it requires special efforts to ensure its containment and prevent its spread through leakage or permeation. For example, there are requirements to have multiple confinement barriers separated by monitored interspaces, and permeation barriers to limit the migration of tritium through the walls of process elements. This is particularly important in high-temperature processes where tritium, like other hydrogen isotopes, can permeate quickly through steel. Because even small amounts of tritium released in this way can present a nuisance to plant operation and become a source of undesirable emission to the environment, containment requirements for these processes exceed normal high-vacuum standards with respect to the leak-tightness to which they must be designed.

The fuel gases must also be handled in accordance with standards applied to safe management of hydrogen - e.g., limiting inventories and guarding against the potential for formation of explosive mixtures which could damage plant equipment. The use of graphite armour for first wall and divertor result in the formation of fine dust in which the fuel atoms are "codeposited". Therefore, ventilation and maintenance equipment must be designed to suppress the spread of these dusts during maintenance activities.

Design depth required during the current ITER Conceptual Design Activities depends on the extent to which uncertainty in the design can affect the credibility of the integrated fuel cycle or the overall machine design. For some elements of the fuel cycle it is only necessary to have a basic conceptual design to understand the impact of the processes on the overall cost, safety, plant layout etc. For others, however, where critical feasibility issues or impacts on major machine components are involved, it is important to understand at an early stage exactly how the process will operate and how it will affect these other machine elements.

For example, plasma exhaust during burn will require high-vacuum pumps for mixtures of D-T and He "ash" with pumping capacities much larger than any high-vacuum system ever built. Not only are the pumps required to be scaled up more than one order of magnitude, they must function in the demanding environment close to the torus. Fuelling systems, such as pellet injectors, must also operate close to the torus and require large amounts of hydrogen propellant gas to be pumped away quickly and processed. Isotopic separation to re-balance the fuel mix involves large flows of tritium, deuterium, and protium, whose inventories must be minimized.

For those systems in which there are critical feasibility issues, activities have been defined as part of the ITER R&D Programme conducted in the laboratories of the ITER partners. All the milestones for these activities have been met to date, lending confidence to the prospect that ITER fuel cycle components can be designed with a solid data-base.

Working together at a distance

Widespread work on design tasks is co-ordinated by Design Unit and Workshops.

At present, the ITER Fuel Cycle Unit has only one "full time" team member - the Design Unit Leader - resident in Garching, who co-ordinates the work of all the contributing countries. For the most part then, design work is performed by the home teams, from which representatives participate in workshops at ITER. To date, three such workshops have been conducted.

The work programme has focussed on the development of design requirements for the process elements in Table 1. The draft preliminary design requirements were prepared using NET-team resources and evolved from discussions and at the first fuel cycle workshop held in the summer of 1988. Since the opportunity for work in close contact is limited (as it is for all ITER Design Groups), a detailed understanding of work assignments and the expected scope of contributions had to be reached with the participants from each ITER Party. At the most recent workshop, held on August 1-11, 1989, participants submitted draft concept descriptions of many of the key elements for the Fuel Cycle design. Based on the extent of work done and the commitment of the home organizations, groups were assigned various tasks to prepare refined versions of these concepts which could ultimately be integrated into a consistent design.

Options

Interim conceptual design includes self-consistent set of options.

Final choices of processes and equipment will be based on developments in progress.

It should be emphasized that the joint development up to this time of an integrated conceptual design, although necessarily choosing which one among various options for processes is to be included in a consistent "picture" of ITER, does NOT represent a final concept selection. In many instances, processes are being developed by the various ITER Parties as part of their ITER R&D contributions which, if successful, could offer advantages over the processes which are presently being being used to carry out the initial design integration exercise. The impact of using these developmental alternatives will also be explored in the course of preparing an integrated design. When the related R&D activities are completed, a decision can be made on the inclusion of these alternatives in the reference design.

Summary

Feasibility of Fuel Cycle is supported by design base.

In summary, as we pass the "half-way" point in the ITER Conceptual Design Activities, a work plan and design process have been evolved which have addressed the special problems of working together at a distance. Critical fuel cycle design issues have been considered, and work is proceeding in the home teams. The culmination of this work at the end of the current phase is expected to show that detailed design is feasible with the design base available and under development. Members of home organizations involved have contributed many innovative ideas and valuable experience. Combining these efforts to date has been challenging and rewarding.

Editor's Note

The tokamak, which is today preeminent among concepts for thermonuclear fusion reactors, has a history that goes back to 1950. Some remarkable facts of the first two decades of the tokamak evolution are given in the following article. At that time, the major tokamak-related events took place in the Soviet Union.

FRAGMENTS OF TOKAMAK HISTORY (1950-1969)

by V. Pistunovich and V. Strelkov,
I. V. Kurchatov Institute of Atomic Energy, Moscow, USSR

In 1950, 19 years before the tokamak concept achieved worldwide recognition, the Soviet scientists A.D. Sakharov and I.E. Tamm originated the idea of a toroidal plasma confinement device in which a large current flowing axially around the torus produces an essential component of the confining magnetic field.

Origin of basic principle

In their first analyses, the scientists envisioned the toroidal current circulating in a ring of solid conductor hung in a toroidal vacuum chamber. From the outset, however, there were serious doubts about the feasibility of suspending a conductor ring and transmitting power to it. This led A.D. Sakharov to suggest a technically more feasible way of achieving the desired compensation of charged particle drift in a toroidal magnetic field. He proposed that this be done "by creation of an axial current *directly in the plasma* using the inductive method." This, then, was the origin of the now well-known basic principle of all tokamak devices.

According to the recollections of I.N. Golovin, who became associated with these studies in the USSR soon after their origination, the results of calculations proving the idea of a D-D fusion reaction in a device with plasma current were first presented by A.D. Sakharov and I.E. Tamm at a meeting on 22 October 1950. Scientific reports on this study were completed in 1951. The main parameters of the D-D reactor were calculated by the scientists on the basis of classic transport coefficients, not taking into account the toroidal effects. They reported the following set of parameters.

Calculated parameters of a D-D fusion reactor

major radius	$R = 12$ m,
minor radius	$a = 2$ m,
magnetic fields	$B_o = 5$ T,
plasma density	$n = 3.10^{20} m^{-3}$ and
temperature	$T = 100$ KeV.

Origin of the name "Tokamak"

The first experimental installation, TMP, was designed to study strong toroidal discharges in a magnetic field. The initiator of its construction was I.N. Golovin, who in 1957 came up with the now-familiar generic name for this type of machine. Golovin suggested an abbreviated but descriptive name - "tokamak," from the Russian words "toroidal'naya kamera" and "magnitnaya katushka", meaning "toroidal chamber" and "magnetic coil." The TMP was put into operation at the Institute of Atomic Energy (IAE) in Moscow in 1955. Its main parameters were:

major radius	$R = 0.8$ m
minor radius	$a = 0.13$ m
magnetic fields	$B_o = 1.5$ T
plasma current	$I_p = 0.25$ MA

Early experiments with TMP installation

Research work at the TMP was carried out by a group of experimentalists led by I.N. Golovin and N.A. Yavlinskij. The vacuum chamber of TMP had been made of ceramic and during the experiments, silicon lines were prominent in the plasma spectrum. Some important experimental results were obtained, including the creation of a quasi-stationary discharge in a strong magnetic field, detection of accelerated electrons (with energy on the order of 0.8 Mev) and intensive oscillations in the plasma. Thus began the long and thorny way of the evolution of the tokamak system.

After the experience with TMP, several other tokamak experiments in the USSR were soon initiated. All were constructed with metallic chambers. The most successful choice of wall material was found to be Cr-Ni stainless steel. The tokamak T-1 was constructed at the IAE in 1958, T-2 in 1959 and TM-2 in 1960.

In the middle of 1957, as proposed by N.A. Yavlinskij and with the active support of the leader of Soviet atomic science, I.V. Kurchatov, design activities were started for a large tokamak, T-3. This machine was designed and constructed in a short time and in 1963 the regular experimental programme was begun. T-3 was for 10 years the largest tokamak in the world (until the completion of TFR in France in 1973).

Results of experiments in T-1, T-2 and TM-2

At T-1, the stability of the plasma column was studied. For the first time, the sharp change of the toroidal column stability was observed at $q > 1$ according to Shafranov criterion. It was found experimentally that the major part of the plasma energy was radiated because of the presence in the plasma of impurities coming from the chamber wall. The design of T-2 enabled improved vacuum conditions which reduced the flow of impurity atoms from the wall into the plasma. Although radiation losses from the plasma remained significant in T-2, the improvements made feasible the investigation of some other mechanisms.

It was discovered in T-2 that the transverse component of the stray magnetic field caused significant displacement of the plasma column and increased the interaction with the limiter, if no correcting coils are used for its compensation. Since these experiments, carried out in 1962, in all subsequent tokamaks, the location of the plasma column has been maintained by special control coils. For example, in T-5 (upgraded T-1), special control coils generated a programmed transverse magnetic field to reduce the effect of the stray field in the presence of a copper screen. A feedback system for the plasma column location control without conducting screen was used first in the world in tokamak T-01, constructed in 1970.

A plasma column that remained macro-stable during the entire duration of a current pulse was created at small installation TM-2 at the beginning of 1962. Electron temperature was about 100 ev. When changing the discharge parameters, a new type of plasma instability was first observed which later came to be called a plasma disruption. This type of instability, which creates critical design problems for a fusion reactor, was later studied in T-3 and T-6. Anomalous plasma resistance was studied at TM-3.

Unparalleled performance achieved

By the end of the 1960s, Soviet experiments had revealed the main features of the behaviour of discharge in tokamaks. For the first time, the deuterium plasma with ion temperature 0.5 kev, density 5.10^{13} cm^{-3} and confinement time on the order of hundredths of a second had been created and thermonuclear neutrons detected.

At an international conference in Novosibirsk in 1968, the Soviet tokamak results were reported. L.A. Artsimovich, who had led the research in tokamaks after the tragic death of N.A. Yavlinskij in 1962, concluded that "the way is open for further increase of plasma temperature to reach eventually the physical thermonuclear level." When some members of the international audience found it difficult to accept the accuracy of the tokamak observations, L.A. Artisimovich invited R.S. Pease to carry out a joint Soviet-British experiment at T-3A.

Confirmation and recognition

A team of British physicists, bringing their own equipment, came to Moscow and made measurements that fully confirmed the outstanding confinement performance of T-3A. This exciting outcome, when it was reported at a conference in Dubna in October 1969, proved to be the start of an avalanche process of tokamak construction. Only a few years later more than 50 tokamaks were in operation around the world.

H.P. Furth's recollection reads: "The experiment has proved previous reports of L.A. Artsimovich's group. Tokamaks are invading the world . . " (translation from Russian).

RECENT RESULTS IN SUPERCONDUCTING TOKAMAK TRIAM-1M

by S. Itoh, Director, Research Institute for Applied Mechanics, Kyushu University, Japan

The high-field superconducting tokamak TRIAM-1M was completed in June 1986. Up to now, we have carried out experiments on confinement of high-field tokamak plasma and the steady-state tokamak operation. In the lower-hybrid current drive (LHCD) experiments, we succeeded in a 3 minutes' tokamak operation as already reported at the last IAEA Conference in Nice (IAEA-CN-50/E-III-2), and recently achieved a 7 minutes' tokamak operation by precise plasma position control with Hall generators. We report here the successful operation of the superconducting tokamak and the newest results on the LHCD experiments.

7 minutes' steady-state operation achieved

Superconducting Tokamak TRIAM-1M

TRIAM-1M is a compact high-field tokamak with the major radius of 0.8 m, limiter radius of 0.12x0.18 m^2, and plasma current of 0.5 MA. The toroidal field magnet system consists of 16 D-shaped Nb_3Sn superconducting coils cooled by a helium bath at 4.5 K. It can produce maximum fields of 8 T at the plasma centre and 11 T at the coil windings which are close to the values required for the international tokamak ITER. After the completion of the machine in June 1986, the superconducting magnet was charged up to the full operating current on the first try without any quenches and achieved a peak field of 11 T at the coil windings as reported at the IAEA Kyoto Conference (IAEA-CN-47/H-II-3).

Magnetic fields close to ITER requirements

The machine has been successfully operated in the working tokamak environment for three years. Plasma experiments have been carried out with the maximum toroidal magnetic field and the superconducting coils demonstrated satisfactory mechnical integrity and thermal stability. This stable and reliable operation of the superconducting machine is encouraging for the design and construction of the next generation superconducting tokamak ITER.

Steady-State Tokamak Operation

LHCD is used for long-pulse operation

In order to make use of the unique features of the superconducting tokamak, we attempted a long pulse operation by LHCD. The RF system for the current drive consists of a 2.45 GHz source with output power of 50 kW and a launcher with four waveguides. The RF power of 20 to 30 kW was injected into the target plasma through the launcher with 110° phasing which gave the maximum current drive efficiency in our experiments. In the first stage, the plasma position control was performed by a conventional method with magnetic coils and low-drift analog integrators and the plasma discharge duration was extended up to 3 minutes by precise gas feed control including hydrogen recycling. However, a new method for the detection of the plasma position was required for the development of the steady-state operation because of the increase of the offset in the integrator output signals. Therefore, we attempted a new plasma position control method based on Hall generators.

Effective use of Hall generators for plasma position control

Four Hall generators were set up in the plasma chamber so as to pick up the poloidal magnetic field produced by the plasma current. Since the output signal of each Hall generator included the component of the toroidal magnetic field according to the accuracy of the mounting angle, we compensated the component with the monitor system of the toroidal field coil current. The most serious problem of using Hall generator in the plasma chamber is the deterioration of the magnetic sensitivity caused by the temperature increase of the generator during a long duration discharge. This deterioration was compensated with the thermocouple mounted near Hall generator. By means of the above-mentioned compensating techniques and the application of low-noise amplifiers, the real poloidal magnetic field due to the plasma current was measured with accuracy better than 5% during long pulse operations, and the plasma current and position were successfully determined.

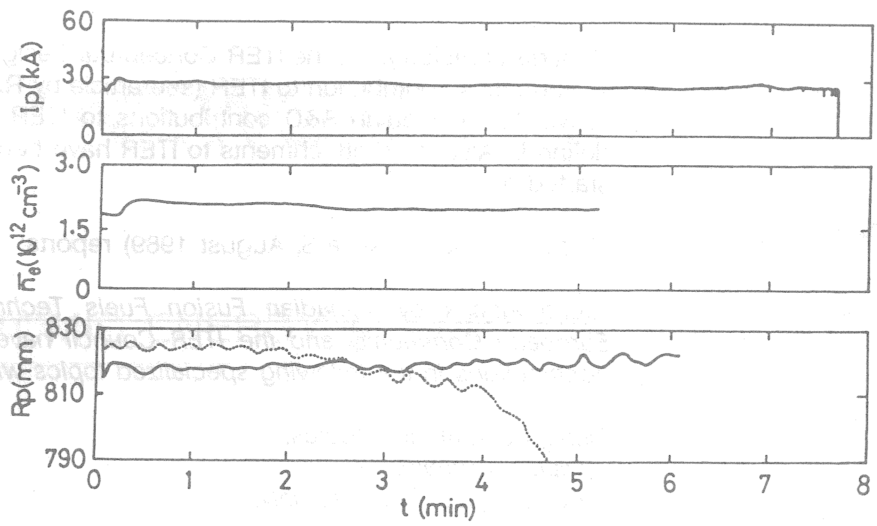

Fig. 1. Waveforms of the longest duration discharge

Characteristics of the longest duration discharge

Fig. 1 shows the waveforms of the discharge with the longest duration of 7 minutes 37 seconds. The plasma with a driven current, I_p of 27 kA was maintained almost constant during this discharge by precise plasma position control with Hall generators as shown by the full line of horizontal position R_p. On the other hand, the dotted line indicates the plasma position measured by the conventional method. It was found that the analog integrators were not applicable for the operation over 3 minutes. The average electron density, $\overline{n}_e$ was also kept constant at the level of $2x10^{12}$ cm^{-3}, and the electron and ion temperatures of the discharge were 1 keV and 0.5 keV, respectively.

We are confident that this success contributes greatly to the development of the steady-state tokamak reactor.

In future, we intend to establish the method of the steady-state tokamak operation and to study the properties of the steady-state tokamak plasma. In addition, we will carry out high-density and long-duration current drive by high-frequency LH waves. An 8.2 GHz RF system with an output power of 200 kW is now being prepared for these LHCD experiments.

CANADA - ITER CONTRIBUTIONS

Canada participates in the ITER Conceptual Design Activities through involvement in Euratom's contribution to ITER (see article by R. Toschi in ITER Newsletter, Vol. 1, No. 2). Canadian R&D contributions to ITER in a number of areas are now defined. Also, staff attachments to ITER have been agreed upon and some have started already.

"Fusion Canada" (issue 8, August 1989) reports:

"Contributions by Canadian Fusion Fuels Technology Project (CFFTP). The European Community and the ITER Council have agreed that design and R&D contributions in the following specialized topics will benefit ITER:

Design contribution topics:
- *Tritium systems.*
- *Maintenance and assembly.*
- *Breeder blanket design.*

R&D contributions are to be made in three of the specific topics identified by the ITER team:
- *Aqueous lithium salt chemistry.*
- *Erosion behaviour of modified graphite in first wall applications.*
- *Fuel systems (fuel handling, isotope separation, tritium extraction, fuel purification)."*

"CCFM Contributions. The scope and exact nature of contributions by Centre canadien de fusion magnétique (CCFM) are currently being formalized. Arrangements may include experimental and modelling work in tokamak plasma impurity transport, and tokamak electrical power system design."

ITER EVENTS CALENDAR - 1989

- Joint Work Session	Garching	1 June - 20 Oct
- Symposium on Fusion Engineering	Knoxville	2 - 6 Oct
- Meeting of Working Party on Ways and Means	Garching	9 - 11 Oct
- ISTAC Meeting	Los Angeles	16 - 18 Nov
- ITER Council Meeting	Vienna	30 Nov - 1 Dec

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