Canadian Fusion Fuels
Technology Project

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INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR - A CANADIAN INVOLVEMENT

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ABSTRACT

An international design team comprised of members from Canada, Europe, Japan, the Soviet Union, and the United States of America, are designing an experimental fusion test reactor. The engineering and testing objectives of this International Thermonuclear Experimental Reactor [ITER] are to validate the design and to demonstrate controlled ignition, extended burn of a deuterium and tritium plasma, and achieve steady state using technology expected to be available by 1990. The concept maximizes flexibility while allowing for a variety of plasma configurations and operating scenarios. During physics phase operation, the machine produces a 22 MA plasma current. In the technology phase, the machine can be reconfigured with a thicker shield and a breeding blanket to operate with an 18 MA plasma current at a major radius of 5.5 meters. Canada's involvement in the areas of safety, facility design, reactor configuration and maintenance builds on our internationally recognized design and operational expertise in developing tritium processes and CANDU related technologies.

INTRODUCTION

The genesis of the International Thermonuclear Experimental Reactor project (ITER) was a Soviet proposal for international cooperation in the development of nuclear fusion tabled at the 1985 Geneva talks between the USA and the USSR. The following year in Reykjavik, the leaders of these two nations took up the theme and included the European Community and Japan in the task of designing a thermonuclear experimental reactor. ITER, latin for "the way", was created under the aegis of the IAEA. Canada was formally admitted in July 1988 under the sponsorship of the European Community and has a permanent chair on the EC team. The joint design team consists of 40 members, 10 from each member nation, in addition other resources in the "home team" of each partner country contribute to the design effort through homework tasks and workshops. A management committee consisting of four directors, one from each of the major partners, runs the operations at the design centre in Garching West Germany, and reports quarterly to the ITER council in Vienna.

The Quadripartite Initiative Committee met in March 1987 and agreed on the project guide lines. In October 1987 the objectives were established and in April 1988 the ITER Council held it's first meeting. From May through September 1988 the first work session was held at the Max Planck Institut Für Plasmaphysik in Garching. The most recent joint work session took place in Garching during February and March 1989. Homework tasks performed by the major partners were reviewed and critical design issues, to be addressed through the balance of 1989, were identified.

There have been a few major changes in the organization of the team since the ITER 1988 summer session. Most of the original team members attended the 1989 winter session. However, each of the partners have added individuals to the list of part-time contributors who will be on-site at some point in the next summer session. As a result, besides the approximately 40 full-time team members, there will be an additional 8 to 12 part-timers from each of the four partners. This does not include contributions from home-team participants which have already been committed.

It should be noted that R. Stasko, the Canadian participant, has been moved from the Basic Device Engineering project group to the Nuclear Engineering Project group. This move, agreed to by R. Toschi, manager of the EC participants, reflects the fact that most of the Canadian R&D and design efforts contribute to the activities of this group.

The project schedule outlines an ambitious program with a 31 month concept and design phase ending in November 1990. Procurement and construction phase commence December 1990 with the start of commissioning in 2002. Operation is divided into two separate phases, physics, from 2002 to 2006 and a technology phase starting in 2008 (after some modification to the machine) and running until 2019.

The terms of reference and objectives for the project are:

1. Define technical characteristics for ITER and do the design work necessary to establish a conceptual design.
2. Define research and development needs, resources and scheduling requirements to realize the design of such a device.
3. Define site requirements for ITER and perform a safety and environmental analysis.
4. Perform specific validating research and development work in support of design activities.

The ITER mission is to demonstrate the scientific and technological feasibility of fusion power. To accomplish this mission, ITER must demonstrate controlled ignition and an extended burn of a deuterium and tritium plasma, ultimately in steady state, as well as demonstrating essential technologies such as superconducting magnets. To be considered successful, the testing phase must also support the design and its extrapolation to commercial power fusion machines. (1)

To put this mission into context with reality the plasma must reach ignition conditions where Q, the ratio of fusion power to plasma heating power, is greater than one. The Lawson diagram (figure 1) shows current experience. There is a long way to go to reach the ITER objective of demonstrating commercial fusion power.

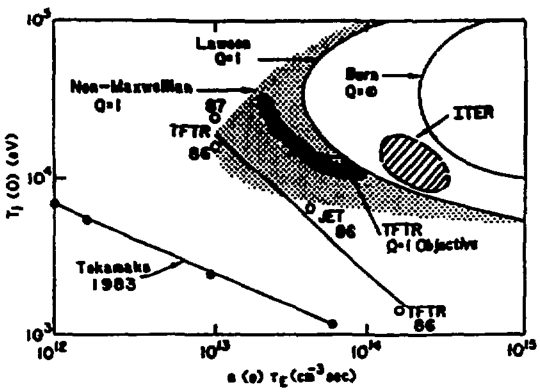

FIGURE 1: LAWSON DIAGRAM

BASIC DEVICE DESIGN CONCEPT

Configuration. ITER is a magnetic confinement fusion device of the Tokamak type. The plasma is contained in a "magnetic bottle" within a vacuum chamber such that the plasma under normal operation will not contact the sides of the chamber. A set of toroidal and poloidal superconducting magnets, (figure 2) keep the plasma suspended in the vacuum chamber and the space between the plasma and the chamber wall acts as a thermal insulator. About 80% of the plasma's energy is carried away by high energy neutrons produced in the fusion reaction. This neutron energy is captured in a shielding blanket in the form of heat. The plasma chamber (19 m diameter by 10 m high) consisting of several segments welded together, (figure 3) is supported inside a set of 16 toroidal superconducting magnets. A set of six superconducting poloidal coils 23 m in diameter is supported outside the TF coils giving rise to an assembly, including support structure, of about 25 m in diameter by 20 m high (figure 4). The centre solenoid portion of the P.F. coil set is self supporting and sits in the hub or hole in the centre of the toroid formed by the inner legs of the T.F. coils.

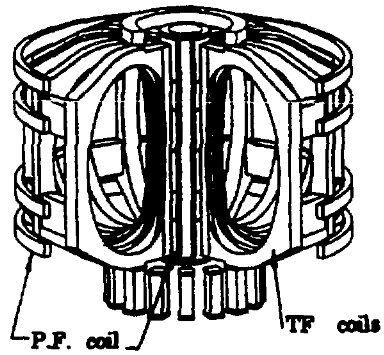

FIGURE 2: MAGNET SYSTEM ASSEMBLY

Current designs call for the tokamak to be housed in a cryostat containing helium for magnet cooling. The cryostat also acts as a secondary containment boundary for the plasma. This entire assembly is set in a concrete cylinder 25 m in diameter and 27 m deep. The cylinder provides structural support as well as biological shielding for external equipment, however this concept is under review and it may be set back about five meters from the machine to improve access for installation and remote maintenance, thus alleviating afterheat problems as well as structural difficulties. (2)

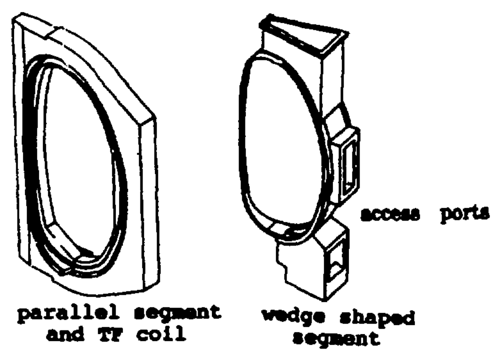

FIGURE 3: VACUUM CHAMBER SEGMENTS

The region between each of the TF coils provides space for 16 access ports, one per sector, in the equatorial plane. Port height is optimized with careful location of the PF coils. Each port is assigned to one or more of the following: test modules, current drive, plasma heating, fuelling, diagnostics, and maintenance equipment.

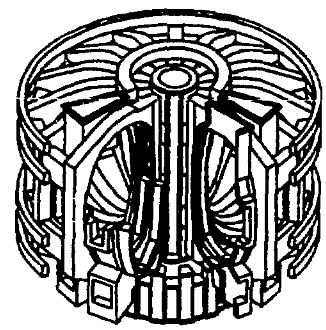

FIGURE 4: BASIC MACHINE ASSEMBLY

The size of ITER is a significant increase over existing machines in order to provide reasonable plasma physics certainty but this represents not only a major undertaking but a large technological challenge (eg, superconducting magnets). A comparison of some of the performance parameters is presented in figure 5.

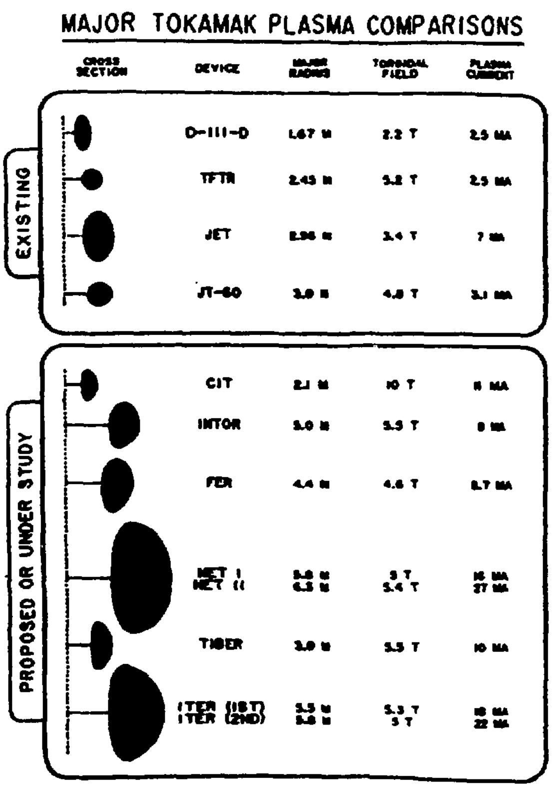

FIGURE 5: TOKAMAK PLASMA COMPARISONS

Maintenance The internals of the plasma chamber consist of test modules, shielding for the magnets, first wall or heat protective surfaces and heat extraction structures. These components are to be maintained by robotic devices which have the capability of in-situ maintenance or removal of components from the Tokamak to a hot cell for repair or disposal. From the point of view of assembly and maintenance, components fall into 3 categories; semi-permanent, medium lived, and short lived. The semi-permanent components will be replaced when they fail if they inhibit the safe operation of the machine. Failure of semi-permanent components implies a major shut down of one to two years. Medium life components such as blankets may be routinely removed to accommodate changes in the machine geometry. The short life in-vessel components such as divertors, first wall tiles, and guard limiters which face the plasma could see frequent damage and need to be "easily" removed to minimize reactor down time. The first wall reference concept specifies a carbon composite armour tile, mechanically attached or bonded to the water cooled heat removal system. The operating load on the first wall is approximately 1.0 MW/m^2 and the ITER engineering test mission requires the equivalent of one full reactor year of operation at this fluence level to prove suitability of tile material. ITER maintenance and design requirements, as listed below, are very ambitious from an engineering stand point.

-ITER should be fully remotely maintainable but with provision for hands on maintenance everywhere possible.

-ITER is to operate in staged operation phases with variable blanket and divertor geometry possible.

-Maintenance of short life and high failure rate components should be possible without moving other components or in any way disturbing the reactor's internal and external environment.

These requirements have a profound influence on both the design of components and their integration into an operable facility. To ensure the robotics are up to the task, representative parts of the machine will be assembled using remote handling devices in a benign environment. A typical magnet installation is shown in figure 6.

Machine Operation There are two operational phases required to achieve ITER's mission. The physics phase has three stages: zero, low, and high activation. During the first and second stages general physics studies of H/D plasmas will investigate energy confinement, particle exhaust, and disruption characteristics and control. The third stage will focus on ignition, controlled burn, and driven operation which will lead into the technology phase. Development of ignited plasma conditions without disruptions, and the characterization of divertor performance for technology phase operation, will be key issues for the five-year 15000-shot physics phase. The physics phase configuration will employ a thin shield blanket allowing for a large plasma cross section and having sufficient volt-seconds to achieve 22 MA current by full inductive operation. With the use of external current drive devices, larger plasma currents are envisioned. At the end of the physics phase, the internals of the machine will be replaced with a configuration compatible with the technology phase mission.

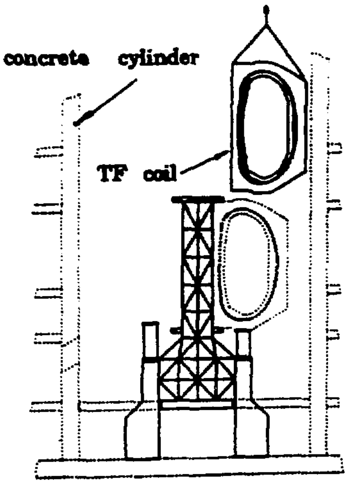

FIGURE 6: MAGNET INSTALLATION

The technology phase will consist of several stages. The first, about three years in duration, will be devoted to concept verification tests of power reactor blanket options, first wall and divertor performance. Assuming the first stage of testing is successful, the technology phase would be extended for about seven years to perform long term blanket testing and proof-of-concept, as well as providing materials damage information. Safety related transient tests would be conducted near the end of the operating period.

SYSTEMS

The myriad of systems required to operate a Tokamak must be integrated so as to minimize complexity and cost, and optimize ease of operation and maintainability. Some of the more controversial systems are considered below.

Fuelling Fuelling for ITER will be gas puffing in the region of the upper divertor. Pellet injection for density, ramp-up and profile control, which are more advanced concepts, might well be considered for later stages of the program. The injector velocity requirements are in the order of 1 to 2 km/s with a repetition rate of 1 to 3 Hz. High speed injectors > 2 km/s may be used if required with the proviso that the technology can be developed. For injection deep into the plasma, higher speeds are being investigated in the USA using compact toroids.
Torus exhaust pumping speeds of 1000 m^3/s or higher at 10^{6} Pa will be required. The reference pumping system calls for compound cryopumps or turbomoloculer pumps.

Current Drive and Heating The current drive and heating systems must satisfy the physics requirements for bulk plasma heating to ignition and steady-state non-inductive operation. In addition, they should provide plasma start-up assistance, profile control and current ramp-up. Given that the technology can be developed, any of the systems proposed could heat ITER to ignition, however the goal is that the same system provide both heating and current drive. Four systems have been proposed; EC (electron cyclotron), IC (ion cyclotron), LH (lower hybrid), and NB (neutral beam injection). Three options were proposed to meet these requirements. In each case, LH drives current in the outer region of the plasma and EC initiates plasma and provide disruption control. Current drive in the centre of the plasma would be provided by either NB, EC, or IC. The preferred option using only LH, and EC is the simplest, the most credible option employs neutral beams as well as LH and EC. These systems are power consumers; for the selected option, up to 280 MW for 100 MW injected into the plasma. A main feature of the most credible option (NB) is its size - much larger than that of the tokamak. The neutralizer, 8 m in length is to be located with unrealistic accuracy in a 40 m long duct. Twelve ion sources per beam, each 4 m in diameter are housed in a concrete bunker 30 m wide, 35 m long and 26 m high. There are 3 such beam lines for ITER each having similar vacuum pumping requirements as the entire torus.

Blankets The tritium-breeding blanket concepts have been narrowed to three options; aqueous lithium salt, solid ceramic, and lead lithium eutectic. All concepts are to be low-temperature low-pressure systems using 316 L austenitic stainless steel for the structural and coolant piping material. For a viable fusion power reactor there is a need to investigate reactor-relevant breeding blanket concepts as well as high-grade heat extraction capabilities of blankets. There is an intent to design full sector blankets with the cross-section and shape of a 14 m long banana. Moving such a component into place and maintaining it remotely will be a challenge.

ITER TECHNOLOGY ISSUES

In the aftermath of the most recent work session (spring of 1989, at Garching) the level of overall design-related activity is expected to increase, as each of the four major participants addresses the significant technological impediments and knowledge shortfalls that will have to be overcome before detailed design commences. These are some of the major issues which will dominate the design homework performed by national home teams, with preliminary results to be reported during the 1989 summer session.

Divertor Technology At present, there is no known divertor concept which will survive the heat load associated with the technology phase of ITER operation. The plasma facing surface will have to survive average heat loads of *5.0 MW/m²* with a peaking factor of two or greater. No design based on the existing materials database for fusion can assure that divertors will survive more than a few days in this environment. Since this has been recognized as a critical technology issue for design of the machine, ITER project management has charged the designers to explore alternate concepts that might still be incorporated into the ITER baseline design. Several options are being developed for presentation and evaluation during the 1989 summer session.

Magnet Reliability This is an important design issue, as the reliability and availability of the large TF and PF coils required for ITER cannot be predicted with sufficient confidence. If any of these superconducting coils requires replacement during the facility operating life, a downtime of several years is anticipated. This could be reduced to several months if replacements are incorporated into the design configuration (ie: in-situ spares) or if they are at least on-site. Establishing detailed failure modes and frequencies can have a large impact on machine cost. At time of writing, no superconducting coils of the required size and current capacities are in operation, and smaller coils have limited experience, or have not performed with the desired reliability. However, the operating experience base at Tora Supra (Cadarache, France) and T-15 (Kurchatov Institute, USSR) is expected to provide design-relevant guidance over the next few years as these superconducting machines begin their experimental programs.

Disruption Forces Forces on in-vessel components resulting from the disruption of a 25 MA plasma current will be in the range of 10-100 MN over a time span of a few milliseconds. At present, designing fastening mechanisms for blanket modules which would survive such forces, yet which would enable blanket replacement, appear problematic. Present concepts involve the use of hydraulic locking wedges or inflatable bladders which will fix the modules firmly in place after remote installation. The vulnerability of such systems to failures, or degradation due to neutron bombardment must be assessed before a reference concept is selected. Designers from all the home teams are addressing this issue.

Tritium Inventory In order to limit offsite doses to 10 rem or less (at a 1 km exclusion radius), the ITER safety group has specified a design target that no system failure mode can result in releases of more than 200 g of tritium (as HTO) from the facility. This is not an insurmountable goal for system designers. For example, the whole tritium systems building is not expected to contain an inventory of more than 350 g and possibly less. Graphite armour tile material can absorb several kilograms of tritium over the operating life of the facility and is presently envisioned as part of the first wall design. Mechanisms will have to be found to limit this uptake, to release it under controlled conditions, or to ensure that failure modes which release part or all of this inventory will leave the containment intact.

Helium Transport Although there is a large margin of uncertainty, physicists have updated calculations on the rate of helium transport from the plasma centre to plasma edge, which indicate that this transport could be slower than originally envisioned. In order to ensure that helium does not poison a self-sustaining fusion ignition, it may be necessary, for example, to increase the exhaust pumping by as much as a factor of ten. Experiments with helium at existing tokamaks may help in clearing up this uncertainty.

CANADA'S DESIGN/R&D CONTRIBUTIONS TO ITER

Design contributions for Canadian home-team participants are still evolving, but are expected to concentrate in the areas of safety system engineering, blanket design, tritium systems, remote maintenance and facility layout. Approximately 8 man years of effort have been identified by the EC as the Canadian component. There has already been significant contribution to the design of the tritium purification systems; to assembly and maintenance of the torus configuration; and to the layout and equipment location of torus systems in the reactor hall.

We have been involved in the design and development of ceramic and aqueous salt driver blankets for ITER. Radiolysis and other chemistry issues for the aqueous salt, and irradiation of a ceramic blanket test module, are our major R&D contributions to ITER.

Design tasks in the tritium technologies include design of structures and air handling systems to maintain tritium leak-tightness. In addition Canada is involved in the preparation of design documents for an aqueous salt blanket tritium extraction system. Other Canadian activities include design of a tritium Isotopic separation system, and design and integration of tritium systems in general.

Canada's expertise in tritium technology and fission plant design will be employed in the development of atmospheric clean-up and processing systems, as well as the handling and disposal of tritiated wastes. The influence of systems containing tritium on plant arrangement and facility design will be another opportunity area.

By hosting atmospheric release studies with other nations, Canada has taken a lead role in the investigation of the environmental effects of tritium. These activities and our tritium dosimetry capability positions us to make useful contributions to the safety studies associated with ITER.

Our CANDU experience in design and operation will be drawn on in areas of engineering assessment, reliability, safety and costing studies.

The need for remote maintenance skills on ITER assures Canada a role both in robotics design and influencing the configuration of the Tokamak. Canada has placed engineers from Spar in key remote handling design roles in both JET and NET. In addition Wardrop Engineering has an engineer stationed at NET developing remote maintenance designs. We are involved in the design concept of an in-vessel vehicle which will perform in-situ repairs, as well as remove components from the Tokamak for ex-vessel repair.

In the critical area of first wall and divertor materials, Canada is providing key data on graphite and modified graphite properties on exposure to plasmas, including tritium retention and outgassing. This work is being conducted at the University of Toronto.

CONCLUSIONS

It should be noted that although Canadian contributions have stemmed from areas of R&D and engineering activities associated with the CFFTP fusion program, contributions from other research and engineering agencies in Canada are anticipated later in the design phase. This process will be furthered by a continuation of the broadening of Canada's technical interests, as well as recognition by the International community that Canada's technical contributions are worthwhile; all the more so as design effort expands during the latter phase of the project.

The ITER project was conceived as a study with the design phase scheduled for completion in 1990. It remains to be seen whether the four parties will proceed with joint construction of ITER beyond 1990. The course of events will hinge on both technical merit and the international political climate. If ITER becomes a real facility, Canada will most assuredly be on the team and make a notable contribution.

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