CANADIAN CONTRIBUTION TO THE EUROPEAN UNION HOME TEAM PROGRAM FOR ITER

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ABSTRACT

Canadian participation in R&D and design tasks for the ITER project is predominantly in the Fuel Cycle, Remote Handling and Safety fields. These tasks are carried out in Canada by Ontario Hydro, research institutes, industry and universities. In addition, Canada provides the services of a number of specialist engineers and scientists in key positions at the three ITER work sites and in the European Home Team. The Canadian contribution, which is coordinated by the Canadian Fusion Fuels Technology Project (CFFTP), forms an integral part of the European Union Home Team program. The key components of the Canadian contribution are described.

INTRODUCTION

The world fusion program is developing the design of a machine called ITER, the International Thermonuclear Experimental Reactor, and carrying out the necessary R&D program. This machine will facilitate the study of the behaviour of hot fusing plasmas under fusion reactor-relevant conditions, and the testing of prototypes of the hardware needed to support these in a fusion power plant. Because of its size and associated cost, and the breadth of the technical challenges, ITER is being carried out as an international project within the framework of a quadripartite (EU, USA, Russian Federation and Japan) program.

Canada has been an active participant in the ITER project since its inception in the mid 1980's. The Canadian contribution to ITER includes both R&D and design tasks, and is provided to ITER as an integral part of the European Union (EU) program. Activities in the Canadian program are focused on the Fuel Cycle, Remote Handling and Safety fields. Key elements of these tasks, and their integration into the overall ITER program, are described below.

FUEL CYCLE

The fuel cycle includes one set of systems that handle the fuel from storage, through fuelling and pumping, to cleanup and isotopic adjustment, and a further set of systems to detritiate aqueous and gaseous waste streams. Only a small fraction of the deuterium-tritium (DT) fuel is burned in one pass through the fusion chamber, so systems are needed to recover and recycle this fuel. As tritium is mildly radioactive (a low-energy beta emitter), extra precautions are needed for worker and public safety.

This work program (Dautovich and Miller, 1995) is devoted to developing the specialized systems and materials required for this fuel cycle. Canada is responsible for the isotope separation system (ISS), water detritiation system (WDS) and atmosphere detritiation system (ADS), and is contributing to the fuel cleanup, fuelling, vacuum pumping, fuel storage, blanket tritium recovery, plasma-facing component tritium, and tritium plant control systems. In addition, a dynamic computer model to study the transfer of tritium between systems has been developed for ITER in Canada.

The WDS and ADS are based on proven CANDU technologies. Issues addressed for the ITER WDS include feed treatment and off-line analytics. For the ADS, module and component sizings were optimised. The other systems in the fuel cycle, however, have required more research to support the ITER design, as outlined below.

Fuel Cleanup System

The ITER plasma exhaust will consist mostly of hydrogen and helium gas, but there are expected to be several percent of various impurities and reaction compounds, including water, CO/CO_2 and hydrocarbons. In order to remove these impurities, permeators will be used. This will produce impurity-free hydrogen to feed to the ISS. However, the removed impurities will contain some tritium. In order to keep releases to the environment at acceptably low levels, the Fuel Cleanup System must detritiate its feed gases by an overall factor of around 100,000,000.

Considerable resources have been devoted by several of the ITER parties to the development of processes to handle this demanding duty. Within the EU program, a close collaboration between CFFTP and the Karlsruhe Research Centre, Germany (FzK) has been instrumental in validation of process components and concepts by testing them in different test loops and under different operating conditions. The processes proposed to meet this target, usually involve two stages of impurity detritiation downstream of the permeator front end. The first process to demonstrate the high detritiation factor required for the impurity detritiation was the Canadian-developed HITEX (High Temperature Isotope Exchange) process. In this process, the impurity gases are mixed with pure hydrogen (protium) in a process called "isotopic swamping". The tritium in the impurities exchanges with the hydrogen gas over a suitable catalyst. The tritiated hydrogen is then separated by a permeator. In experiments in 1994, detritiation factors of up to 1,000,000 were achieved in a single HITEX stage, much better than the 10,000 or so required in this step (Miller et al., 1995 and 1995).

In the 1995-96 experiments (Rodriguo et al., 1996, 1996 and 1997), tests were made with other feed compositions more representative of expected ITER conditions, notably containing water, or CO/CO_2. In runs with moist feed gas mixtures carried out using an AECL Platinum-on-silicalite catalyst, with the critical process lines trace-heated to 80C, the target detritiation factor of 10,000 was achieved in less than one hour, with incremental detritiation still taking place, but at a lower rate (See Fig. 1).

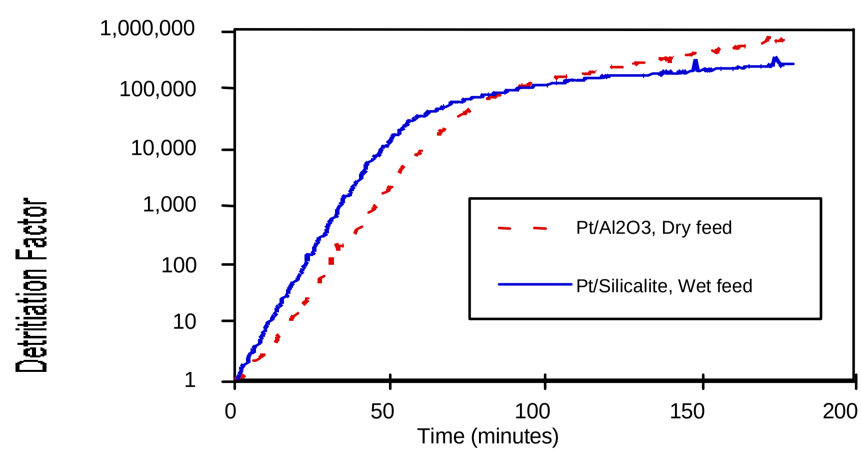

Fig. 1 Detritiation Factor vs Time for HITEX Tests

In 1996 the test facility at AECL Chalk River Laboratory underwent a major rebuild, to enable different combinations of the two impurity detritiation steps to be tested in series, and a direct comparison of the performances of different concepts to be obtained.

Experiments were also carried out at FzK in their Fuel Clean-up test loop, modified for HITEX-like operation. These experiments confirmed the importance of minimizing moisture retention on the process lines and in the catalyst, and of making the process loop simple, eliminating “dead” volumes that the gas does not sweep effectively. In related work, an alternative isotope swamping process, called PERMCAT, in which the hydrogen gas flows countercurrent to the impurity gas, has been proposed by FzK (Glugla et al., 1996). With the appropriate conditions, very high detritiation rates should be achieved in a once-through mode. High tritium tests in the upgraded facility at Chalk River were carried out and confirmed the expected good behaviour, with detritiation factors of up to 1,000,000.

Isotope Separation System

The Isotope Separation System (ISS) in ITER is based on Cryogenic Distillation (CD), the same process as is used in Ontario Hydro’s Darlington Tritium Removal Facility (DTRF). However, while the ITER system will process less hydrogen, it must produce four product streams from feeds that can vary significantly in both isotopic composition and flow rates. Moreover, the ITER ISS tritium feed rate will be up to several hundred grams per hour compared with the DTRF feed rate of up to 1g/h of tritium. It is essential to reduce the inventory of both tritium and total hydrogen in this system in order to minimise radiological and hydrogen explosion hazards respectively, and to save costs.

Ontario Hydro and CFFTP have developed a number of improvements to cryogenic distillation technology over the past several years. Some of these were used in a novel compact cryogenic distillation system installed on the TFTR (US) tokamak in support of its tritium operation. This TFTR system was put into routine operation in 1996, producing >99% pure tritium that could be recycled back into the reactor as fuel, rather than the previous process wherein tritium was converted to water and shipped off-site as waste. This was the first time that fusion fuel had been recycled in a major fusion experiment, representing a major milestone in the development of individual systems and their integration into a closed cycle.

For ITER, our work has consisted of developing innovative design solutions to handle the ITER requirements, coupled with improved experimental understanding of the behaviour of columns and the associated distillation packings (Kveton et al., 1995, Sood et al., 1997). Early in 1996, the throughput requirements on the ITER ISS were increased by a factor of 10, presenting us with the challenge of accommodating this increase in throughput without a significant increase in tritium inventory. Using a combination of superior column packings, optimized operating conditions and improved cascade configuration, we have accommodated the increased flow rate with little overall change in tritium inventory (Bellamy et al., 1995, Woodall, 1996 and Woodall and Bellamy,1997).

Figure 2 illustrates how the tritium inventory in the final column of the ITER ISS has decreased (for a given throughput) as various design and experimental results have been incorporated. Current experiments are aimed at quarter-scale tests to confirm a total tritium inventory of 200g., now predicted for most conditions of ITER ISS operation. An upgrade of the test facility, including a four-fold increase in refrigeration power was carried out in 1997 to accommodate these test runs.

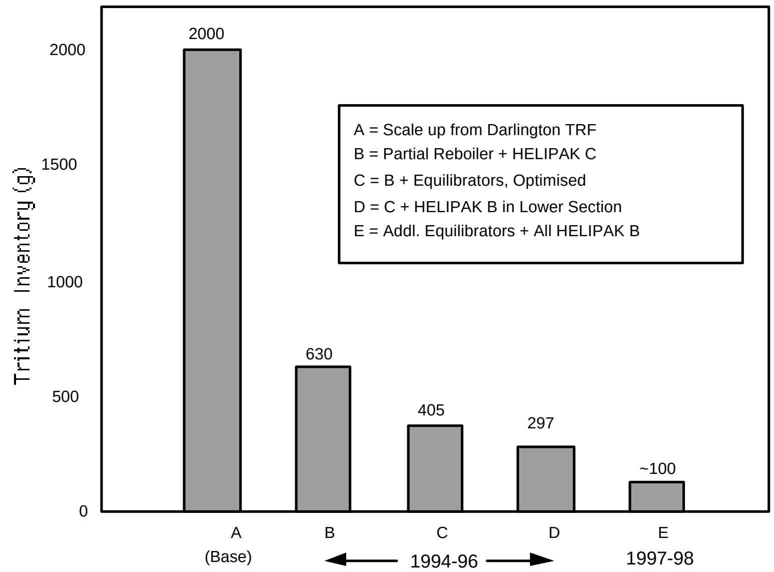

Fig. 2 Isotope Separation System - Inventory Reduction in Final Column (based on throughput of 200Pa.m^3/s.)

CT Fueling

Compact Toroid (CT) fuelling involves injecting self-contained toroid-shaped plasmas containing hydrogen fuel into the main tokamak plasma at high speeds. This method offers a deep fuelling capability unattainable with other techniques. Deep fuelling should allow better plasma density control and lower tritium throughput, the latter contributing significantly to reducing in-vessel tritium inventory. A proof-of-principle experiment is under way on the Tokamak de Varennes at the Centre Canadien de Fusion Magnetique. In order to produce cleaner CT plasmas with greater penetrating power, several modifications were implemented in the last year. These included a larger diameter accelerator, improved gas valves, oil-heated electrodes, and an increased capacity power supply (Raman and Gierszewski, 1997 and 1997). Based on these experiments, a concept for ITER, where greater CT mass and velocity are required, is being developed.

Plasma Facing Components

The potential for high tritium inventories is present in the plasma-facing components. Work at Canadian universities has focused on understanding the interaction of hydrogen isotopes with first wall materials based on carbon, tungsten or beryllium. Recent work includes systematic measurement of the erosion of graphite at the very low plasma energies now expected in the ITER divertor, and characterisation of removal rates of codeposited hydrogen in amorphous carbon layers by reaction with moist and dry oxygen. (Haasz et al., 1997, Thompson and Macaulay-Newcombe,1997 and 1997).

Integrated model of ITER fuel cycle

The ITER fuel cycle is dynamic. Flows of gases between process components vary significantly between burn and dwell, and even during steady burn, as various pumps are cycled, and the fuelling rate, composition, and method are adjusted to optimise plasma conditions. These variations in flow rates, and the pulsed nature of operation, give rise to substantial net transfers of tritium inventory between systems. It is important to understand these dynamics as the process systems must be designed to accommodate them.

The CFTSIM code has been developed to track the fuel cycle dynamics on time scales of single burns (1,000-10,000s.) to multiple shot campaigns. It models the hydrogen gas flows, inventories and isotopic compositions for the key tritium-handling processes, including the fuelling system, neutral beams, plasma and first wall, vacuum pumps, fuel cleanup system, isotope separation system and fuel storage system. The code builds on previously developed dynamic models for several systems in the Ontario Hydro's FLOSHEET and DYNSIM codes.

In 1996, the first version of this code was released for ITER use. It contained a detailed model of the ITER ISS, but relatively simple models of the other ITER fuel cycle systems. It is presently being upgraded to include, among other changes, more detailed fuelling, neutral beam, and fuel storage models. Also, an improved torus module is being developed under a collaboration between ITER and the US Home Team.

REMOTE HANDLING

Divertor Duct Maintenance (L-7)

Canada's largest remote handling contribution to ITER is the design and construction of a prototype divertor duct vehicle for remote maintenance (Millard and Blevins 1997). This is one of the key subsystems of the full-scale divertor replacement mock-up being prepared by the EU-HT. The divertor duct vehicle will remove plugs to access the divertor, and prepare the site before and after divertor replacement. The vehicle is equipped with an assembly for moving an 8 tonne door plug, an assembly to install and remove rail sections, and an industrial manipulator plus tool rack for remote bolting, cutting and welding operations. The prototype was built and tested in Canada by Spar Aerospace, and then shipped to the European test facility in autumn 1997.

The Duct Vehicle System can unseal and remove the large port plugs to permit the servicing of the divertors. To facilitate these tasks the vehicle deploys a rail system, cuts and re-welds port plug lip seals and divertor cooling piping. The system was designed using 3D CAD and kinematic simulation to analyse the duct operations and the design requirements followed by design prototyping and optimisation. This methodology allowed the most critical features of the duct and vehicle system design to be finalised relatively quickly, enabling the design, fabrication, and acceptance tests to be completed in 18 months.

The divertor handling vehicle system is designed to unbolt and remove the cryostat closure plate, deploy sections of radial rail, unbolt, cut and remove the vacuum vessel closure plate and deploy a second rail section to access the divertor cassette. The duct vehicle system comprises four elements: a vehicle, a manipulator, a rail handler, and tools.

The duct vehicle is a remotely operated, rail-mounted, electric powered truck with a 10 tonne lift capacity, designed to pitch a payload forward 2 degrees and backwards 47 degrees. The lift mechanism incorporates a self-aligning pick-up interface, which allows the vehicle to engage either a duct door, a manipulator arm, or a rail adapter (used for lifting sections of removable rail). The pitch axis serves two functions, aligning the payload with the duct structure, and pitching the payload back so as to clear obstacles as the system moves back down the duct. Debris is sucked away from the tool face during cutting operations and is collected in a cyclone separator located in the manipulator frame. The vehicle houses an air pump and a water pump to operate the tool connector plate and support the manipulator tools operation. Lights and cameras are mounted on the vehicle and manipulator for visual aid.

The manipulator is a modified electric industrial robot with 6 degrees of freedom, a 45kg payload capacity at full acceleration and velocity and a repeatability of less than 0.5 mm. A tool exchange plate mounted on a force/moment sensor enables tools to be exchanged on the manipulator and provides power, data, video, gas and coolant services to the tools (two bolting tools, a lip seal cutter, and a gripper). An automotive bolt tool and controller were modified to provide automatic bolt head alignment, avoid transmitting torque through the robot, keep the bolt captive and improve operator awareness during the bolting operation. The vehicle with rail adapter places rail sections 1.3 metres long weighing 1.5 tonnes.

The control equipment is housed in two separate cabinets and a remote operator console.

The operator interface consists of a graphical user interface (GUI), translational and rotational rate hand controllers, a hardware switch and power status panel, along with video monitors and a speaker system. The hardware switch panel contains power enable/disable switches for the vehicle, manipulator and tools, and an emergency stop switch. The control system supplied as part of the Canadian scope interfaces with the supervisory divertor maintenance control system supplied by our European partner on this task, ENEA (Italy).

Bore Tooling System

Bore tooling systems developed in collaboration with Comex Nuclear, Marseilles has being modified to suit L-7 project requirements. Canada is supplying the tool deployment system, and its controls based on Steerable Tubular Extendible Member (STEM) technology. The STEM is mounted at the open end of a pipe and the bore tools are inserted and positioned to a prescribed location by the STEM to perform either cut or weld operations.

Other Remote Handling Activities

Canada has also developed a design for the maintenance systems for the main cryopumps, and participated in a study to identify the commonality of requirements and design features between the maintenance tasks for the Divertor (accessed via the lower set of radial ports, as described above) and the Blanket Segments (accessed through the equatorial ports), in order to rationalise equipment and procedures.

Other Canadian contributions include testing of new concepts for radiation-hardened electronics (Hiemstra, 1996), radiation-hardened optical-fibre databuses, 3-D vision systems, optical-fibre strain gauges.

SAFETY

Canada is supporting the ITER Safety and Environment work, including activities related to radiation protection (occupational radiation safety), effluent source terms and emissions, and plant safety.

Occupational radiation safety

The work in radiation protection is to demonstrate that occupational safety has been adequately addressed in the ITER design by reviewing and assessing operational and maintenance activities and estimating worker tritium exposure for these activities. In addition, a Radiation Protection Plan for workers, with guidelines for access control, radiation monitoring, implementation of ALARA principles, and health physics, is being prepared.

The first part of the Radiological Occupational Safety Assessment has been completed (Kalyanam et al, 1995). A qualitative occupational radiation exposure assessment for operations/maintenance for the Hot Cell, Heat Transport System, Cryopump and Divertor replacement was carried out.

CFFTP provided a specialist to the ITER San Diego site to assist JCT in outlining a radiation protection program and in writing the occupational radiation safety section (Volume VI) of the Non-Site Specific Safety Report (NSSR). This report is a key document in licensing ITER. Modifications necessary to meet the local regulatory requirements will be incorporated in due course to generate a site-specific version of the document. Specific documents on Radiation Access Zoning, ALARA guidelines for use in the design process, and radiation protection planning were prepared.

Source terms and emissions

Effluent assessment continues to focus on updating the tritium effluent quantities, based on a room-by-room, component-by-component analysis during different ITER operational states, including maintenance and postulated accident scenarios. A tritium map is charted which indicates possible tritium releases and concentrations in working premises. Two reports to determine preliminary estimates for tritium releases have been issued (Kalyanam and Natalizio, 1997). The estimated tritium release is within the targets established for ITER.

A methodology document for determination of leakage from Heat Transport Systems was prepared. A preliminary escape estimate for ITER, scaled according to number of components and coolant loop conditions was made (Kalyanam et al, 1995). Results indicate an overall leakage rate of the order of 10 Mg per year of tritiated cooling water could be expected. This leakage is to the vault area, where it will vaporise into the room atmosphere; the room ventilation system will be designed to recover this.

Plant Safety

The task covering the Safety Assessment of Fuelling and Vacuum Pumping Systems was completed and a final report for the event sequence analysis submitted. The work was focused on the identification of initiating events and event sequence analysis . In addition, an assessment of potential explosion and fire hazards in the tritium plant and in the fuelling systems is being conducted and mitigation strategies proposed. The methodology of determining hydrogen release and the modelling of an explosion was outlined. A preliminary analysis of a representative case has demonstrated the adequacy of the methodology.

CONCLUSIONS

The Canadian work is being carried out through CFFTP by Ontario Hydro, research institutions, industry and universities. Many of the tasks build on technologies originally developed in support of CANDU generating plants, and thus make essential technology available to ITER in a cost effective manner which is synergistic with ongoing CANDU operations.

These system- and component-based tasks are complemented by the assignment of Canadian specialists to key positions in the European Home Team and the ITER JCT Design Team. Canadian assigned staff are presently, or have been, working on all three of the ITER Joint Work Sites, and in all of the technical fields described above.

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KEYWORDS

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