FUSION FUEL BLANKET TECHNOLOGY

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I.J. Hastings, P. Gierszewski
Atomic Energy of Canada Limited
Canadian Fusion Fuels Technology Project

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I.J. HASTINGS

Atomic Energy of Canada Limited - Research Company
Chalk River Nuclear Laboratories
Chalk River, Ontario KOJ 1JO

and

P. GIERSZEWSKI

Canadian Fusion Fuels Technology Project
2700 Lakeshore Road West
Mississauga, Ontario L5J 1K3

ABSTRACT

The fusion blanket surrounds the burning hydrogen core of a fusion reactor. It is in this blanket that most of the energy released by the nuclear fusion of deuterium-tritium is converted into useful product, and where tritium fuel is produced to enable further operation of the reactor. As fusion research turns from present short-pulse physics experiments to long-burn engineering tests in the 1990's, energy removal and tritium production capabilities become important. This technology will involve new materials, conditions and processes with applications both to fusion and beyond. In this paper, we introduce features of proposed blanket designs and update the status of international research. In focusing on the Canadian blanket technology program, we discuss the aqueous lithium salt blanket concept, and the in-reactor tritium recovery test program.

INTRODUCTION

There are many possible fusion reactions. The simplest is the reaction between deuterium and tritium, two isotopes of hydrogen. Most of the reaction energy is released as high energy neutrons (14 MeV rather than the 3 MeV neutrons typical of fission reactions), which penetrate about one meter into most materials before being completely stopped.

Consequently, the first function of the blanket is to surround the hot reacting hydrogen with a medium that stops these neutrons and converts their energy into a useful product. The nature of this medium depends on the product. The applications of fusion are primarily electric power and fissile fuel production. However, there are other ways to take advantage of the neutron and plasma energy available, including isotope production (e.g., ^{60}Co), synfuel production, nuclear waste burnup and space propulsion. In the simplest application, neutron energy is converted to heat, removed by the coolant and used to make steam for electricity. Since tritium is not naturally available, the second major function of the blanket is to produce tritium. This is accomplished by use of lithium and neutron-multiplying materials. Lithium has a high probability for absorbing an incident neutron and transmuting into tritium. Neutron-multipliers (e.g., beryllium, lead, zirconium) have a high probability of releasing two neutrons upon absorbing one high-energy neutron. The net result is that the fusion device consumes deuterium, lithium and neutron multiplier.

There are presently a variety of blanket concepts under consideration. The details vary with the specific application and the reactor conditions. The most important reactor conditions are the incident neutron flux, reaction burn time, lifetime irradiation fluence and magnetic field strength. These conditions vary according to the nature of the device (e.g., experimental test reactor, compact power reactor), and between device types (e.g. magnetic-confined plasma in a tokamak configuration, inertially-confined reaction driven by high-power lasers). There is considerable choice of materials for structure, coolant, breeder (lithium-bearing material) and multiplier (1). Table 1 lists materials of most interest. Although not all combinations are sensible (e.g. liquid lithium breeder with water coolant), there are still a large number of possible combinations with their own unique characteristics.

TABLE 1: PRIMARY BLANKET MATERIAL OPTIONS.

Breeder	Coolant	Structure	Neutron Multiplier
Liquid metal	Liquid metal	Austenitic steel	Beryllium
Solid ceramic	Helium	Ferritic steel	Lead
Aqueous salt	Water	Refractory alloy	
Molten salt		Martensitic steel	
Solid metal			

Self-cooled blanket concepts use a single medium as both breeder and coolant. Since this medium must contain lithium to breed tritium, the primary options are liquid lithium, liquid lithium-lead (a eutectic), lithium salts dissolved in water, and molten lithium salts. These blankets offer good heat transfer (the neutron energy is deposited directly in the coolant) and mechanical simplicity (Figure 1). However, liquid metals interact strongly with the magnetic field present in many fusion device concepts, which sharply inhibits their motion. They are also relatively chemically reactive and require high temperatures, an electrical conversion efficiency advantage but a safety disadvantage. Water is a good coolant and is not sensitive to magnetic fields, but must have a high dissolved salt content for adequate

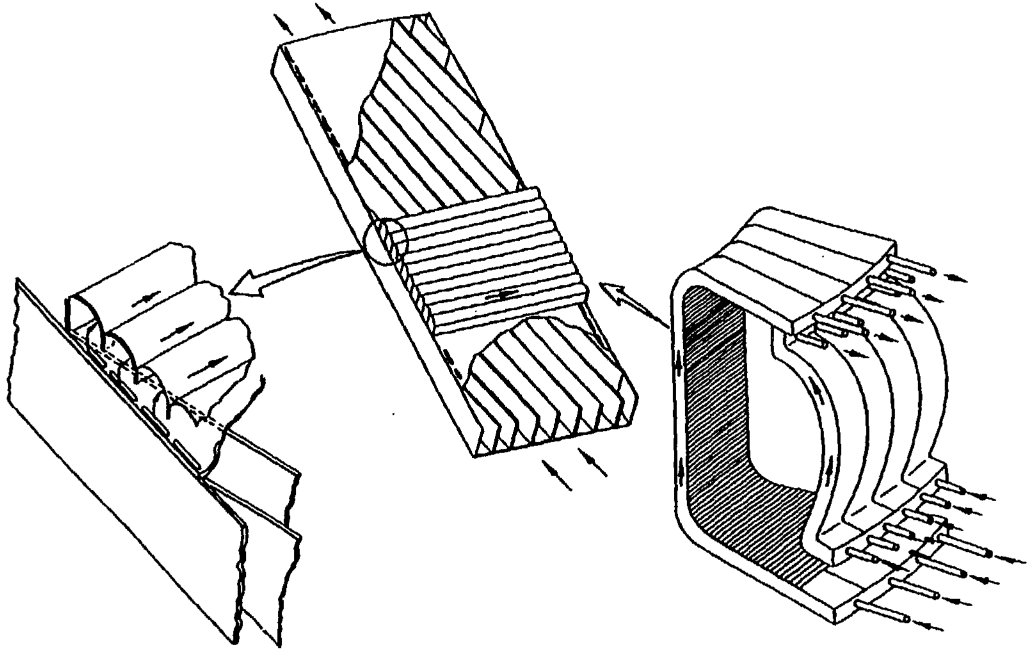

FIGURE 1 SELF-COOLED LIQUID METAL BLANKET SECTION FOR A TOKAMAK REACTOR (1).

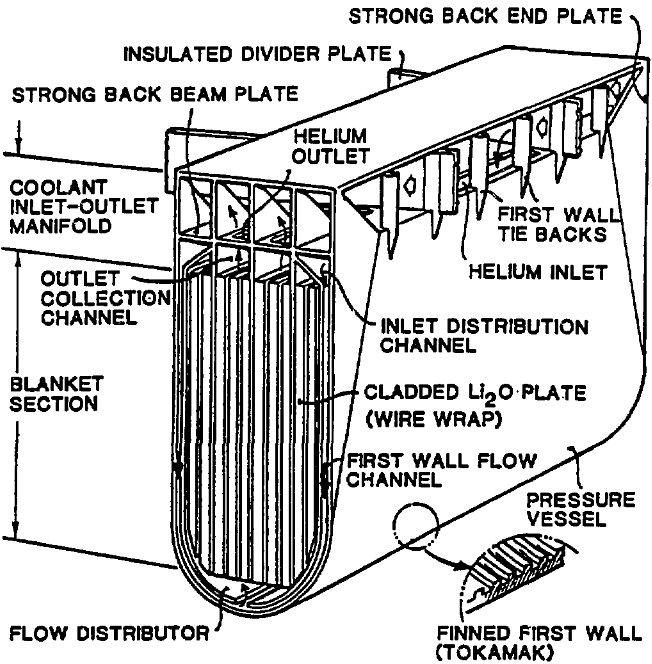

FIGURE 2 HELIUM-COOLED SOLID BREEDER BLANKET MODULE (1).

tritium production (2). Tritium recovery from the aqueous system is well established compared to that for other breeding materials, but requires isotope separation, so is more expensive. This is a recent concept, and a preferred solution has not been identified, although several have been proposed. Molten salts could provide adequate cooling and high temperatures (for good thermal efficiency in electricity production) even at low pressures. The primary problems of lithium-bearing salts is their high melting point (360°C for $LiF-BeF_2$), and chemistry-related concerns.

The alternative to self-cooled blankets is to separate the two functions of breeder and coolant and try to find an optimum material for each function separately. This increases the flexibility in materials choice and reduces the risk of tritium loss across primary coolant heat exchangers. However, the blanket becomes mechanically more complex, (Figure 2) with thousands of small coolant channels running through the breeder, and the extraction of tritium from the breeder becomes more difficult. The options for the breeder are liquid lithium or lithium-lead metals, or solids such as lithium oxide or lithium aluminate ceramic, or Li_7Pb_2 alloy. There are many possible solids, although those with adequate tritium breeding and tritium release are much more limited. The tritium would be extracted by running a hot helium gas purge through the porous solid bed, or by slowly circulating the liquid metal, so it would not be affected by magnetic fields.

The conditions in inertially-confined fusion are characterized by short pulse operation, modest vacuum and no magnetic fields, quite different from magnetic-confined fusion devices. Blanket concepts for inertial fusion devices (such as laser or particle beam driven fusion) are typically based on 'waterfalls' of liquid metal or solid pebbles to provide a continuallyreplaced 'first wall' which protects the primary structure from fatigue due to the highly pulsed reaction.

Since many of the required materials, conditions and technologies are new, there are important issues that need to be resolved. These have recently been identified and characterized in the FINESSE study (3), and are summarized in Table 2.

TABLE 2: CRITICAL ISSUES FOR BLANKET TECHNOLOGY

Tritium fuel self-sufficiency
Structural behaviour under irradiation
Tritium extraction and control
Magnetohydrodynamic effects (liquid metal blankets)
Materials compatibility (liquid breeder blankets especially)
Breeder/structure thermomechanical stability (solid breeder blankets)

Achieving tritium fuel self-sufficiency requires that the achievable tritium breeding ratio be greater than the required breeding ratio. The latter suffers from uncertainties related to performance of plasma (e.g., tritium fractional burnup) and hardware. The uncertainties in the achievable breeding ratio are more serious for solid breeder than liquid metal blankets, but can be resolved if enough neutron multiplier, for example, beryllium, is used.

The major issues specific to liquid metal blankets are magnetic field effects, particularly on fluid flow, pressure drop and heat transfer; and materials compatibility (e.g., corrosion of structural materials by liquid metals, and reactivity with air and water). In addition, structural response, tritium extraction and control, and general failure modes are not well understood at present. A program to resolve these issues will include a number of small and large loop experiments. The issues and testing needs for solid breeder blankets differ from liquid metal blankets in several important respects. Firstly, there is a much wider range of possible materials to consider. Secondly, the influence of geometry on the primary uncertaities is not as important. Thirdly, the behaviour and the issues are dominated by the effects of irradiation. These issues, in addition to tritium self-sufficiency and structural behaviour, are tritium recovery and inventory in the solid, and thermomechanical behaviour of the solid breeder and cladding.

INTERNATIONAL STATUS

At this early stage in the development of blankets, the focus is on understanding fundamental behaviour and materials. There have been no full blankets constructed or tested under conditions resembling fusion reactor conditions. However, there have been and are a number of significant international blanket experiments.

In the solid breeder area, for example, there are several irradiation experiments underway, including at FFTF (US), SILOE (France), Petten (Netherlands) and in Japan. These are similar to the CRITIC experiment at Chalk River Nuclear Laboratories (CRNL) (see later), although generally producing less tritium and focussing more on the low temperature end of the data. There is a significant closed-capsule irradiation experiment that just started at FFTF (US) in 1986, where about 20 solid breeder materials with different characteristics from various countries and fabrication processes are being irradiated under identical conditions to high lithium atom burnups. The results of this test will provide important comparative data between the materials, as well as on the effects of high burnups in general. A follow-up experiment is being discussed, which would irradiate fewer and larger pellets under more reactor-like conditions. Canada is participating in the assessment of this advanced tritium recovery test, and may contribute breeder materials and tritium testing technology.

BEATRIX, the International Breeder Exchange Matrix under the auspices of the International Energy Agency, has been a successful example of international co-operation on solid blanket technology. Canada, via CRNL, is a full partner in BEATRIX; the US, EEC and Japan are other participants. A recent matrix document showed 19 individual experiments. Canada was the first to carry out testing under BEATRIX, on lithium aluminate from CEA (Saclay, France). The US is providing lithium oxide for the major CRITIC-I tritium recovery test in NRU, discussed later.

In the liquid breeder area, there are several large loop experiments underway to explore magnetic field effects and corrosion. There are plans for tritium extraction tests, including in-reactor capsules (Europe), but as yet these experiments have been limited.

The development of structural materials which can withstand long irradiation by fusion neutrons, have high-temperature strength, and have neutron activation, has been a major element of the fusion technology programs. Recently, alloy compositions of ferritic steels have been irradiated to the equivalent of two years of reactor life (out of an anticipated four-year life in the innermost components) with excellent resistance to irradiation damage. Special austenitic steels and refractory alloys are also under advanced testing.

CANADIAN BLANKET TECHNOLOGY PROGRAM

Program Organization

Canada has considerable experience in nuclear and aerospace technologies that are relevant to fusion. There is a National Fusion Program that supports research and development in two major areas, Canadian Fusion Fuels Technology Project (CFFTP) and the Tokamak de Varennes. CFFTP is a joint effort by the Federal government, the Ontario government and Ontario Hydro, and is developing fusion-fuels related systems and services. Since the blanket is the primary fuel production system in a fusion reactor and contains critical issues and interesting technological challenges, development of blanket technology has been an important program for CFFTP. Most research and development in this area has been performed by Atomic Energy of Canada Limited at Chalk River Nuclear Laboratories (CRNL), under a cost-sharing agreement with CFFTP.

The general areas of most interest are solid breeders, since the issues (fabrication, tritium recovery, irradiation behaviour) are generically similar to issues that had been successfully addressed as part of the CANDU fission reactor program. Some additional effort has considered blanket concepts and other potential areas for Canadian research. In particular, liquid breeders were identified as an interesting alternative to solid breeders. One successful result of this effort is the aqueous lithium salt blanket concept which is currently enjoying considerable international interest.

The present major goals of the Canadian blanket technology program are: 1) develop and test an optimized breeder ceramic material and fabrication process; and 2) develop and test the aqueous lithium salt blanket concept, for tritium production in an Engineering Test Reactor. The first area is being addressed by breeder fabrication development and tests of the critical tritium recovery issues. The latter was originally explored at a conceptual level, but is now turning to detailed experiments and analysis. It is possible that both these projects will lead to major Canadian contributions to international experiments in the next five year period, and later in the 1990's be used directly in an Engineering Test Reactor.

Fabrication Development

CRNL's Advanced Ceramics group operates laboratories that are well equipped for the small-scale development of advanced ceramic processes and materials. The group also makes use of other facilities and expertise at Chalk River, notably for work involving colloid chemistry, mechanical testing, chemical analysis, nuclear magnetic resonance (NMR) and non-destructive testing.

Of the potential solid lithium breeder candidates, two have been selected for further consideration: the oxide and the aluminate. Lithium oxide has the highest lithium density and so is the most desirable from a breeder standpoint. The main problem with the oxide is its high affinity for water vapour; any processing or handling must be performed in a glovebox with good atmospheric control. Lithium aluminate is one of the most preferred compounds, with the oxide, at the present time. It does not have as high a lithium density but is easy to work with in a normal laboratory environment. Effort has been focused on this compound to make rapid progress in the development of fabrication concepts.

Most solid breeder materials have been made in the form of pellets pressed from powders. In the case of aluminate, the powder is commonly made by decomposing the carbonate in the presence of alumina. Recently, advantages have been cited for fuel in the form of microspheres (spherepac), and several approaches by other investigators have been pursued in that direction. CRNL is investigating powder techniques for fabricating pellets, and both powder and sol-gel approaches for making microspheres. Rotary agglomeration (4) is a promising novel method of preparing microspheres from powders whereby a suitable powder is agglomerated into microspheres by tumbling. Figure 3 shows microspheres about 2 mm in diameter produced by this technique at CRNL. Sintered densities are

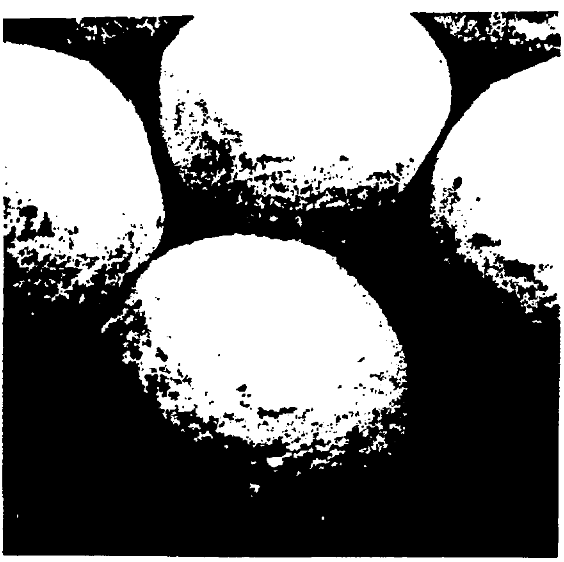

FIGURE 3: LITHIUM ALUMINATE MICROSPHERES 2MM DIAMETER FABRICATED BY ROTARY AGGLOMERATION.

65-70% of theoretical. This approach to making microspheres is unique within the breeder blanket fabrication community; preliminary results have been encouraging.

Irradiation Facilities

The major irradiation facility is the NRU research reactor at CRNL. The reactor has a capability for two types of irradiation tests to evaluate solid breeder ceramics. The focus so far has been on lithium aluminate and lithium oxide.

Unvented Capsule Tests

The unvented capsule tests, designated CREATE (Chalk River Experiment to Assess Tritium Emission), are those from which tritium release information is obtained after the irradiation is complete, and the capsule is removed from the reactor. Information can also be obtained on the form of tritium released, as a function of capsule material and sweep gas composition.

Maximum sample size is 2 cm diameter and 15 cm long. Typically, samples weighing 50-100 mg are cut from sintered pellets of the ceramic for irradiation. Each sample is vacuum-annealed in a quartz tube, and sealed in the tube for irradiation without further exposure to air. $LiAlO_2$ samples are annealed at 400°C and 3×10^{-2} Pa for 1 h; Li_2O samples at 600°C and 3×10^{-2} Pa for 6 h. Samples are then irradiated for 48 h at an average flux of 7×10^{16} $n.m^2.s^{-1}$ and an estimated temperature of less than 100°C. The maximum flux available is 4×10^{18} $n.m^2.s^{-1}$ (thermal) and 7×10^{17} $n^2.s^{-1}$ greater than than 1 MeV). The free tritium recovered at room temperature is measured, as well as the isothermal tritium release at the postirradiation test temperature. Both tritiated water and reduced tritium are determined. The tritiated water may include T_2O and HTO, and the reduced tritium T_2 and HT, but for simplicity only HTO and HT are used in this paper to refer to these tritium forms. The HTO is removed in the first ethylene glycol bubbler and the sweep gas then passes through an ionization chamber, which provides on-line monitoring of the HT released. A second measurement of the HT is obtained by passing the sweep gas through a CuO bed to convert the HT to HTO, and another set of ethylene glycol bubblers to remove the HTO. The total integrated release of both HT and HTO is determined by analyzing the bubbler solutions using liquid scintillation counting. The time dependence of only the HT release is obtained from the ionization chamber readings.

Both He and He-1% H_2 are used as sweep gases at a flow of 0.5 L/min. The He is purified by passing it through a hot titanium bed; the He-1% H_2, by passing it through a Deoxo unit and a molecular sieve drier. The oxygen and moisture contents of the purified gas are less than 1 µL/L. Extraction vessels constructed from quartz, stainless steel, Inconel-600 and nickel are available. Tests are performed at up to 600°C for times to four hours. Tritium remaining in the ceramic after annealing is recovered by dissolving the sample in 6 N HCl, neutralizing with 6 N NaOH and distilling the resulting solution. Tritium in the distillate is determined by liquid scintillation counting.

Full details of earlier unvented CREATE tests have been given elsewhere (5-7).

Vented Capsule Tests

The vented capsule tests, designated CRITIC (Chalk River In-Reactor Tritium Instrumented Capsule) (5) permit continuous on-line monitoring of the tritium release from the ceramic during the irradiation, by passing a sweep gas around or through the ceramic and into an analysis train. Since fusion reactors will probably use sweep gas to recover the tritium in the same way, the experiment attempts to model a miniature segment of a blanket. Figure 4 shows a diagram of the CRITIC assembly. A sample size 4 cm diameter by about 10 cm long is possible in the current capsule.

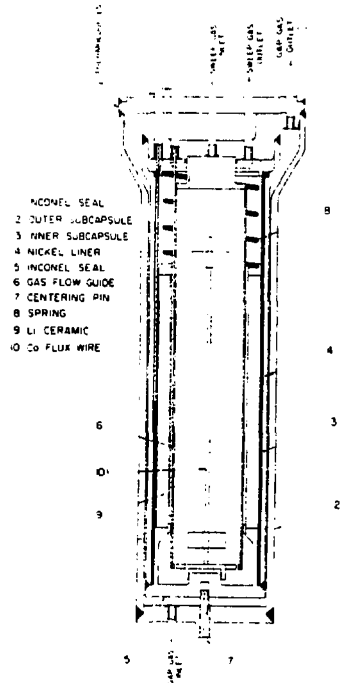

FIGURE 4: CRITIC-I CAPSULE.

Tritium is generated in the ceramic by neutron-induced fission of lithium, and migrates to the ceramic surface. Flowing sweep gas collects the tritium released from the lithium ceramic and passes through a tritium analysis system to determine the release rate and tritium form. The ceramic is typically in the form of sintered pellets, with 10-20% porosity. The capsule provides approximately uniform ceramic temperatures to facilitate analysis of release data; a small radial temperature gradient of about 50°C enables calculation of thermal conductivity of the ceramic. The temperature is adjustable between 400 and 900°C, which covers the range of expected operation in a commercial reactor, by varying the composition of an insulating gas layer (gap gas).

The form and rate of tritium release may be affected by factors such as surface impurities on the ceramic (particularly H_2O and CO_2), the composition of the sweep gas, and the materials chosen for the capsule walls. Therefore, the ceramic must be carefully fabricated and loaded into the capsule to minimize contamination by air. The sweep gas composition can be varied (He to He-1% H_2) during the course of the experiment to determine the influence of composition. The capsule will be fabricated from nickel-based alloys, rather than stainless steel, to reduce container influence on the form of the tritium. Nickel-based alloys also provide more strength at high temperature. Finally, to minimize condensation of HTO in the sweep gas lines, the gas lines are trace-heated to about 100°C between the reactor and the tritium analysis system.

It is possible to measure important fundamental parameters so that the behaviour of other blanket assemblies can be inferred: diffusion, desorption, and heat transfer coefficients can be calculated as a function of temperature. In addition to on-line tritium analysis, gamma spectroscopy monitors the release rate of trace quantities of other radioactive species. These are expected both from neutron activation and from fission of uranium impurities in the ceramic. A moveable spectrometer is located at the glove box containing the tritium analysis system, and a portable spectrometer is available adjacent to the gas line exit ports from the reactor. Other instrumentation include thermocouples, on-line flux monitors, and integrated flux monitors. Analysis of the gap gas permits measurement of the permeation rate of tritium through the Inconel capsule wall.

There are differences in neutronics between CRITIC and a blanket in a future fusion reactor. In CRITIC, tritium production will occur almost exclusively from thermal neutron captures by ^6Li. In a fusion reactor, tritium will be generated from captures, by both ^6Li and ^7Li, of neutrons within a wide energy range, up to 14 MeV. In both cases, helium will also be produced, and the stoichiometry of the ceramic will change. The average tritium production rate per gram of ceramic in a blanket will not be far different from the rate in CRITIC. Also, in a breeder material in a fusion reactor, displacement from energetic neutrons will be larger than in CRITIC. Near the first wall of a fusion reactor, the displacement damage rate will likely be larger by two orders of magnitude. Near the blanket rear, they will be comparable.

Accelerator Facilities

It is impractical to test all possible candidate materials for fusion reactors by irradiating with neutrons because of the cost in building high flux, 14 MeV, neutron facilities. An alternate and logical approach is to determine the fundamental and engineering properties of various materials, using particle beams (e.g. protons, heavy ions or electrons). This data base, with theoretical calculations, is important for optimum scoping selection of blanket materials.

The main accelerator facilities at CRNL available for implanting heavy ions in materials are 70 kV and 2.5 MV mass separators. In target chambers on these accelerators there are cryopumping facilities that enable low contamination rates to be maintained in the region of the target. This is achieved by surrounding the sample by a large cylinder cooled to approximately 20K during the implantation. Using various heating and cooling stages, the sample temperature can be varied from 35-770K. Facilities are also available for sweeping the ion beams over the target, thus resulting in uniform implantation over areas up to approximately 1 cm^2,

A cryogenic target chamber for light ion irradiation (proton, deuteron, He) has been constructed and incorporated into the beam line of the tandem accelerator super conducting cyclotron (TASCC) at CRNL. An energy range of 15-22 MeV has been chosen because it best simulates the energy spectrum of the primary knock-on atoms produced by 14 MeV neutrons. The most common problem associated with ion irradiation, especially at cryogenic temperatures, is beam heating. To minimize this, the specimens are immersed in a low pressure helium cooling gas (approximately 0.1 Pa). To contain the gas in the chamber, indium wire gaskets and two thin (5 μm Harvar foil) beam windows, one for beam entrance and one for exit, were soldered to the chamber wall. Two wire-wound heaters located at the cold finger provide heating for postirradiation annealing. The ion beam is swept over a length of 2.5 cm using a sweeping magnet upstream.

RESULTS

CREATE Tests

Table 3 summarizes the eight CREATE tests performed so far at Chalk River. Previously (6,7), the effect of the oxygen activity of the experimental system on the form of the tritium recovered from $LiAlO_2$ and Li_2O was demonstrated. With He-1% H_2 or a stainless steel extraction tube, oxygen activity was low, and tritium was recovered primarily as HT. With pure He and more chemically inert extraction tube, tritium was recovered primarily as HTO. There was, however, significant variation in the HT/HTO ratio for samples tested under similar conditions indicating the importance of possible material and system impurities on the results. Data from French $LiAlO_2$ under the BEATRIX Program show results comparable with those for previous CREATE tests (7,8).

Table 4 gives preliminary data from the most-recently analyzed unvented test, CREATE-VI. The Li_2O pellets were received from JAERI (Japan) under the BEATRIX program and ranged in density from 67 to 85% of theoretical. The experimental conditions, percentage of free tritium, and percentage of HTO in the total tritium desorbed at the chosen temperature are summarized in Table 4. Comments on the effect of experimental conditions on the amount of free tritium and the HT/HTO ratio of the desorbed tritium are included in Table 4.

The HT release rate from the samples cut from Pellet #3 (73% TD) is variable for the runs carried out at the same temperatures (two at 400°C and two at 600°C) making it difficult to determine the effect of temperature on the HT release from these particular samples. However, the HT release from the samples from Pellet #5 (80% TD) and Pellet #7 (85% TD) does demonstrate the effect of temperature on the release rate. In Pellet #7, 100% release is observed in 200 and 30 min at 500°C and 600°C, respectively. Further analysis of the release rate will be carried out once work to determine grain and pore size, and microstructural analysis, is completed. Further details of CREATE-VI are given elsewhere (9).

TABLE 3: CREATE CLOSED CAPSULE TESTS AT CHALK RIVER

CREATE	CERAMIC	ORIGIN	YEAR	POST-IRRADIATION ANALYSIS
I	$LiAlO_2$	ANL	1985	Test T release and analysis system
II	$LiAlO_2$	ANL	1985	Quartz, stainless steel and nickel container; HT released with He+1%H_2; HTO released with He except with stainless steel
III	$LiAlO_2$	CRNL	1985	Effect of powder processing on T release
IV	$LiAlO_2$ Li_2O	ANL	1985	Confirm CREATE-II results; Effect of breeder material on T release
V	$LiAlO_2$	France	1985	T release at temperatures from 500-600 °C; 165 kJ/mole activation energy for T release
VI	Li_2O	Japan	1986	Measure T release
VII	Li_2O	ANL	1986	Test material to be used in CRITIC-I open capsule irradiation experiment
VIII	$LiAlO_2$	Japan	1986	Irradiated, awaiting analysis

ANL - Argonne National Laboratory
CRNL- Chalk River Nuclear Laboratories

TABLE 4: PRELIMINARY CREATE VI RESULTS FROM JAERI LITHIUM OXIDE

Li_2O Material	Sample #	Vessel*	Temp. (°C)	Sweep Gas	Flow Rate (mL/min)	Free Tritium (% of total)	Desorbed Tritium % HTO	Comments
Pellet # 2 67% TD	13	I	600	He/1%H_2	500	18	8	High % of free tritium for all of these samples. Effect of adding H_2 to sweep gas in reducing %HTO is noted. With pure He, larger % of HTO was expected although ~50% has been observed previously for similar conditions.
	15	I	600	He/1%H_2	500	17	7	
	16	I	600	He	500	9	58	
	17	I	600	He	500	58	52	
	14	Q	600	He	500	4	63	
	18	Q	600	He	100	26	45	
Pellet # 3 73% TD	3	Q	400	He	100	18	25	%HTO very low for experimental conditions for both runs at 400°C. The expected %HTO was obtained at 600°C (sample 12).
	9	Q	400	He	100	2	5	
	10	Q	600	He	100	<0.1	total T determined	
	12	Q	600	He	100	0.5	73	
Pellet # 5 80% TD	1	Ti	600	He	500	11	79	High % of free tritium. Residual moisture in vessel or system may have contributed to large % of HTO for initial run. The two subsequent runs with Ti gave only a very small amount of HTO even with pure He.
	4	Ti	600	He	500	52	8	
	2	Ti	500	He	500	15	3	
	3	Q	600	He/1%H_2	500	30	12	Large % of HTO at 450°C and 400°C, opposite to results obtained for samples of Pellet # 3 under similar conditions. Low % of free tritium compared to other runs with same sample.
	5	Q	450	He	100	<0.1	74	
	6	Q	400	He	100	2	99	
Pellet # 7 85% TD	23	I	600	He	100	10	29	Generally low % of free tritium compared to samples from other pellets. Except for Sample # 23, expected % of HTO was obtained. This may be due to a "conditioned" Inconel vessel as a result of its previous use.
	19	Q	600	He	100	0.5	93	
	21	Q	550	He	100	<0.1	85	
	20	Q	500	He	100	0.3	92	
	24	Q	600	He/1%H_2	100	7	4 —	

* I = Inconel, Q = Quartz, Ti = Titanium

CRITIC Test

The CRITIC *in-situ* tritium release experiment began 1987 January 30 at CRNL. This test places 100 g of Li_2O into the NRU thermal reactor. The Li_2O is 1.53 wt% Li-6, 30 mm ID, 40 mm OD annular pellets of 91% theoretical density in a 100 mm stack. The ceramic was fabricated by Argonne National Laboratory under the BEATRIX (IEA International Breeder Exchange Matrix) program. A six-month irradiation of the heavily-instrumented test rig is planned at 500-900°C, varying the He-H_2 sweep gas, with on-line HT/HTO measurement. Lithium oxide conductivity and tritium permeation from the capsule will also be measured. The test is distinguished by the care taken to observe the tritium release form, using a high tritium production rate (4 Ci/d), heated purge lines (100°C), nickel alloy materials, and the fabrication and capsule loading procedures to minimize H_2O and CO_2 contamination.

The first weeks of irradiation have focussed on slowly drying the Li_2O ceramic while also obtaining tritium release information; the full test matrix will examine T behaviour from 400-900°C in detail. Slow dryout is necessary to prevent possible cracking or breakup of the ceramic. Towards this end, reactor startup was extended over a 5 h period. Since the ceramic had been dried out-reactor (in the capsule) to 300°C, little moisture was measured until the ceramic exceeded that temperature. At full power, when the ceramic temperature ranged from about 400°C to 450°C, the sweep gas dew point had risen to approximately 36°C (200 vppm) from -73°C (2 vppm). In the period to 1987 February 05 this decreased to -56°C; by 1987 February 13, ceramic temperatures had reached 650-700°C and the dew point was -72°C, close to the projected value. To date, the experiment has operated as designed. One vanadium self-powered flux detector stopped working before insertion, but a second detector and the cobalt fluence wires provide sufficient backup. Forty-one signals are being recorded on hard disk on an independent data acquisition system. For backup, 12 signals are also being recorded on a multi-pen chart recorder, and NRU Operations personnel manually record all the key signals once per shift.

Tritium appeared within 3.5 h of startup. We are now measuring 30-40 Ci/m^3 of tritium in the sweep gas stream (flow rate 100 cc/min) with peaks up to 70 Ci/m^3. About 85% of the release so far is HTO; 15% is HT. We expect this to change to a larger proportion of HT when the ceramic is fully dry, and when we use He-H_2 sweep gas. The test is providing valuable data (10) to assist future blanket design studies.

Fundamental Support

In support of the tests of tritium release from engineering materials, more fundamental research is being pursued to explore the basic tritium behaviour in solid breeder materials. Perturbed Angular Correlation techniques have been applied to $LiAlO_2$ and Li_2O. With further development, this should be suitable for observing the nature of defects in solid breeder ceramics. The Elastic Recoil Detection and Nuclear Reaction Analysis methods for profiling hydrogen isotope profiles in near surface regions (about 1 micron) have been developed and applied to irradiated Li_2O and $LiAlO_2$, and to the capsule liners from these breeder irradiation experiments. Linking the fundamental tritium data with those from the CRITIC and CREATE tests will allow development of a model for application over a wide range of conditions.

Aqueous Lithium Salt Blanket

We are also participating in the development of an aqueous lithium salt blanket concept. Figure 5 shows the concept schematically. The advantages of this blanket include its mechanical simplicity (the water is both breeder and coolant), and use of conventional materials and operating conditions (e.g., water-cooled fission reactor technology), and the excellent heat transfer ability of water (2). A commercial high-power density fusion blanket based on this approach is being evaluated by a U.S. TITAN Reversed-Field Pinch design team led by General Atomic, with CFFTP participation. A particularly attractive near-term application for this blanket concept is to provide *in-situ* tritium generation for Engineering Test Reactors. A key question is the most attractive tritium recovery and control methods for reactor-scale conditions.

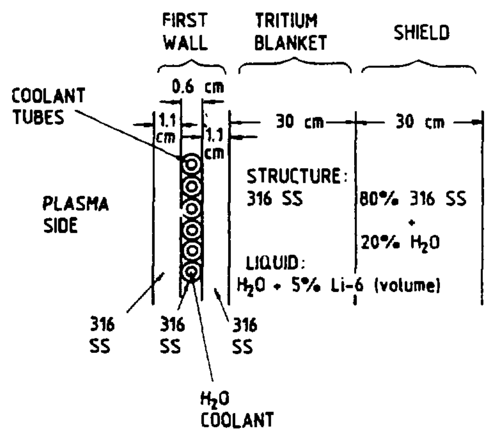

FIGURE 5: SCHEMATIC OUTLINE OF AQUEOUS LITHIUM SALT BLANKET.

For near-term Engineering Test Reactors (12), the blanket would have a simple shield-like geometry, and operate at low temperatures and pressures (80°C and 0.2 MPa). A thin 30 cm blanket would allow a larger plasma volume during the startup phase but still provide, for example, a 40% tritium breeding ratio with 50% plasma coverage. Furthermore, tritium production and contamination of the first wall/blanket/coolant could be delayed until satisfactory plasma operation had been achieved, and then started by the addition of lithium salt to the water coolant and the installation of the tritium recovery system without any in-vessel modifications. This concept is presently being actively considered within the US TIBER and European NET design groups.

Particular projects recently started include a careful assessment of tritium recovery and control systems, and conceptual design, and basic tests of possible solutions with materials they might be exposed to. This blanket concept is of particular interest to Canada since it relies extensively on CANDU technology and is within Canadian industrial capabilities.

SUMMARY

The fusion blanket is a critical component of a commercial fusion device. It surrounds the reacting hydrogen and converts the energetic products into useful products, as well as providing fuel.

The Canadian program has an important element of blanket technology development because of its application to areas of Canadian expertise, and the potential for spinoffs. A particularly relevant background is the CANDU system and the associated expertise and research capabilities of Atomic Energy of Canada Limited (particularly CRNL in advanced ceramics, complex irradiation testing and tritium technology), Ontario Hydro, Canadian universities and industry.

In the next five year program, areas where further advances are expected include the development of unique solid breeder ceramic materials and fabrication processes, the testing of Canadian and international breeder materials in unique Canadian testing facilities, the development and testing of the aqueous lithium salt blanket on a prototypical scale, and the development of particular projects such as developing liquid metal experience and addressing lithium isotope separation.

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