Fusion Materials Issues

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Abstract
Progress toward demonstrating the scientific feasibility of fusion energy is strongly governed by materials constraints. The evolution of an economically and environmentally acceptable design for power-producing fusion reactors will be subject to still further materials constraints. Three critical materials areas are addressed: 1) fast neutron damage. 2) tritium breeding, and 3) plasma-materials interactions. Canadian R & D activities in fusion materials are reviewed.

Résumé
Le progrès pour démontrer la faisabilité scientifique de l'énergie de fusion nucléaire est fortement contrôlé par les limites imposées par les matériaux. L'évolution d'une conception économique et acceptable pour l'environnement de réacteur de puissance à fusion sera le sujet d'autres limites encore imposées par les matériaux. Trois aspects critiques des matériaux seront examinés: 1) dégâts causés par les neutrons rapides, 2) génération de tritium, 3) interaction de matériaux / plasma. Les activités canadiennes en R et D sur les matériaux pour la fusion nucléaire sont examinées.

Introduction

Materials is the queen technology of any advanced technical system. The economics eventually depend upon the materials, the reliability depends on the materials and safety depends upon the materials. I assure you that before we are through with fusion, the physicists will give way to the materials engineers as being the leading lights of fusion. ... Yours is the key without which fusion power will remain forever technologically feasible, but practically useless. [1]

E.E. Kintner
Director of U.S. Fusion Program 1975–81

Few people appreciate the pivotal, indeed controlling, role which materials constraints play in the practical application of scientific ideas. Focus is generally directed to the principles of physics or chemistry which underlie the application. Thus, for most people, the essence of steam power is the expansive force exerted by matter changing state from liquid to gas. The immense task of transforming scientific principles into functioning systems is left to the engineers. Engineers, therefore, are very much aware of the fact that for every man-year of effort which went into understanding the principles of two-phase H_2O, a decade of effort was required on ferrous metal development.

So it has been with all of mankind's successful applications of science. If aircraft were still made of wood and fabric, of how much utility would the principles of aerodynamics be? Jet and rocket propulsion has utterly – and not entirely for the best – changed the face of the world we live in; these propulsion methods are conceptually simple, but quite unrealizable without high-temperature alloys, refractory liners, ablative coatings, etc. Microelectronics has possibly had an even greater impact, and while the physics involved is certainly not trivial, the materials purity and modification requirements of microelectronic components are spectacular and controlling.

Fusion energy has been termed the most difficult technical challenge ever undertaken. And this just for the scientific phase! The materials and engineering challenges of fusion have only begun to be addressed. From the analogy with other technical applications, however, we can safely assume that the materials challenges of fusion will be unprecedented. A look at a magnetic fusion reactor design supports this expectation. In close proximity – only a metre or two apart – occur temperatures higher than the centre of the sun: the core of the fusing plasma at a hundred million degrees; and temperatures lower than in deep space: the superconducting magnet dewers. The fusion environment is not only thermomechanically extreme but the materials are also required to withstand intense, high-energy neutron irradiation. The front-line com-

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ponents additionally suffer an intense plasma-surface interaction due to the plasma directly contacting them at about one million degrees. Many of the materials must not only withstand this harsh environment for long periods, but must carry out exotic functions as well, such as conducting electricity without resistance, or transmuting lithium into tritium – requirements which severely limit the freedom of materials choice.

Essentially, every component of a fusion reactor requires materials development if fusion is to provide economically and environmentally acceptable power. Many of these developments, fortunately, will be pursued independently of fusion, as they involve problems common to other applications. In this brief survey the focus will be on materials problems peculiar to fusion. Some of these are looming as obvious go, no-go problems; others, more prosaic perhaps, could still quietly kill off fusion as being too expensive or too complicated to attract electrical utilities.

Three Critical Materials Problems for Fusion

Among the materials problem areas which are unique to fusion, and where the achievement of solutions will be critical to the success of the entire enterprise, three stand out: 1) fast neutron damage, 2) the tritium breeding blanket, and 3) plasma-materials interactions.

Fast Neutron Damage

Eighty percent of the DT fusion reaction energy is carried off by 14 MeV neutrons. These neutrons are highly penetrating and leave the plasma zone unobstructed, only stopping deep within the surrounding solid-structure 'blanket.' In the process of slowing to thermal energy, the fusion neutrons cause *bulk* (as distinct from *surface*) damage to the first-wall lining, the vacuum structural materials, the tritium-breeding material, the magnetic coils and their support structures, insulators, windows, coolant channels, etc.

At the atomic level the damage takes two principal forms [2]:

1. During elastic collisions momentum is transferred to the lattice atoms, causing them to be displaced, thus creating voids and interstitials; also, the knock-on atoms themselves have very high velocities, causing displacement cascades.
2. Nuclear transmutations occur, such as (n, α) and (n, p) reactions, which result in the formation of gas within the lattice (helium and hydrogen), while simultaneously the elemental composition of the substrate is changed.

In principle, these damage effects are not unique to fusion, of course, since they also occur in fission systems. The fusion neutron spectrum, however, is much *harder*, i.e., more energetic, than the fission one since the 14 MeV neutrons predominate with only slight moderation. Shown in Figure 1 are neutron spectra for various sources [3]:

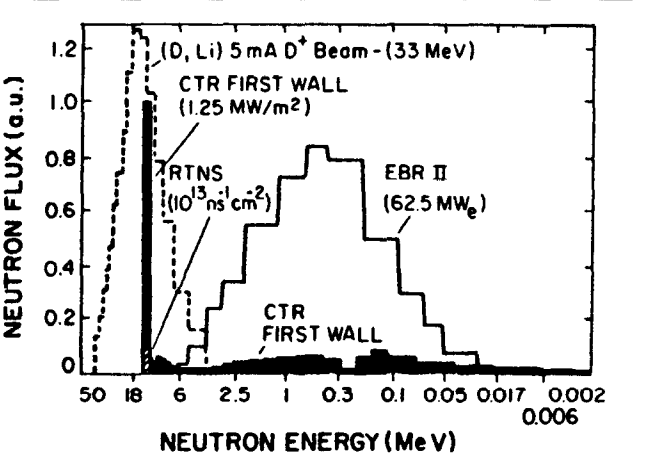


Figure 1: Typical neutron spectra for various nuclear facilities [3].

1. CTR (controlled thermonuclear reactor) first wall – (calculated) spectrum at the fusion reactor first wall for a total power loading of 1.25 MW / m^2 (for economic reasons, a total power loading of ≳1 MW / m^2 is required for a fusion reactor; for DT fusion, 1 MW / m^2 of total power corresponds to 4.4 × 10^{17} (14 MeV) n / m^2 s, and a total neutron flux density about ten times greater).
2. EBR II – a (U.S.) experimental fission reactor used for nuclear materials studies.
3. RTNS – a Rotating Target Neutron Source located at Livermore, California, which employs an accelerated 400 keV D$^+$ beam on a tritiated (titanium) target to produce 14 MeV DT neutrons.
4. (D, Li) – (calculated) neutron spectrum for a proposed D-Li stripping source, using a 33 MeV D$^+$ beam breaking up on a liquid lithium target. The neutron spectrum covers a wide energy band centred at ~16.5 MeV.

As indicated, the fusion neutron spectrum is much harder than the fission one resulting in [2, 3]:

1. higher energy primary knock-on atom, PKA, spectra (Figure 2), causing more lattice displacements (the high-energy knock-ons at ~10^6 eV doing most of the damage);
2. more nuclear transmutations (many transmutations have threshold energies too high to be significant for fission neutrons).

Examples of displacement damage and gas production rates for various materials are given in Table 1 for the fusion spectrum. In brackets are some gas production rates for the EBR-II fission spectrum. One may note the high displacement rate, indicating that each wall atom will be displaced from its lattice position many times each year. The gas production rates in atomic parts per million per year are also very high for fusion.

Synergistic processes may be important, in which case the ratio of gas production to displacement is also relevant. Table 2 indicates that this ratio differs greatly between fission and fusion neutron spectra. The

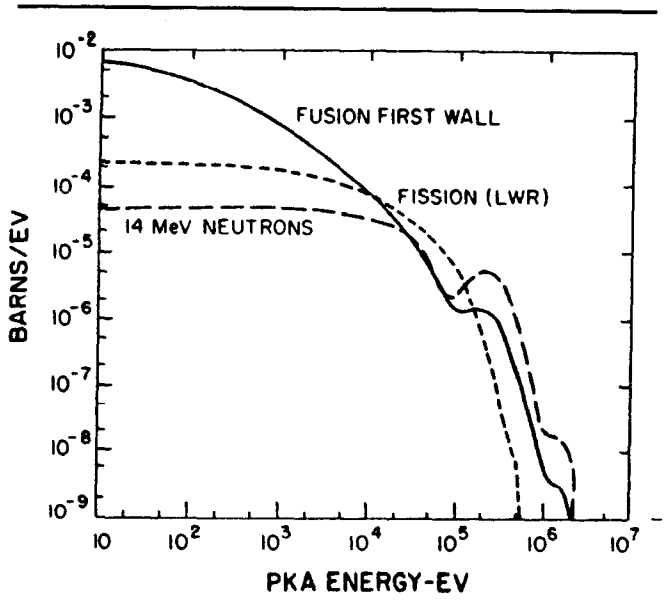


Figure 2: Primary knock-on spectra for copper in various nuclear facilities [3].

elemental composition of the wall changes rapidly (Table 3).

These damage processes occurring at the *microscopic*, or atomic level, then result in a myriad of *macroscopic* materials problems [2, 3]:

1. *Swelling*. Lattice vacancies precipitate into voids causing swelling; the gases – particularly the insoluble helium – also cause swelling due to bubble formation.
2. *Sintering* in some materials reduces available void space, causing contraction.
3. *Growth*. Carbon has enormous advantages as a first wall material (see below), but unfortunately suffers from strong neutron-induced growth, leading to run-away elongation at radiation loadings of 10–20 dpa.
4. *Embrittlement*. Stainless steel suffers virtually total loss of ductility by 100 dpa and 6,000 appm helium. Long before this point, however, the ability of a steel vacuum vessel to maintain ultra-high vacuum integrity over a surface of ~1000 m^2, and subject to thermal cycling, will have been lost.
5. *Creep*. Many of the structural components, such as the vacuum vessel, are subject to high stress and high temperature, resulting in plastic deformation over long periods: creep. The creep rupture life of stainless steel is reduced 50% by neutron irradiation. Even a small plastic deformation will make component disassembly and replacement difficult or impossible.
6. *Fatigue*. Pulsing of the magnets and vacuum system 'works' the metal, inducing fatigue and potential failure. The role of radiation in this process is little understood.
7. *Induced radioactivity*. While DT fusion 'burns clean,' i.e., the fuel ash itself is not radioactive, the neutron bombardment of the reactor walls induces radioactivity via transmutations. In the case of stainless steel, the radioactive (structural) waste to be disposed of at the time of de-commissioning the reactor would not be enormously less than the radioactive (fuel-ash-plus-structural) waste from an advanced fission system (liquid metal fast breeder) (Figure 3a), and a strong incentive therefore exists to develop more exotic metallurgies, such as that of vanadium (Figure 3b), for fusion structural components.

These neutron-related materials problems are daunting, and it remains to be demonstrated that solutions can be found which will permit economical operation of fusion reactors, with reasonable maintenance efforts and acceptable environmental impact.

In light of the seriousness of this problem, considerable thought has gone into potential solutions. The first step has been obvious for many years – although little action has yet occurred – namely, the develop-

Table 1: The Displacement Damage Rate and Gas Production Rates in Typical Fusion Reactor Materials, Based on a Time-Averaged Neutron Wall Loading of 1 MW / m^2

Alloy	*Displacement damage rate dpa / year*	*Helium production rate appm / year*	*Hydrogen production rate appm / year*
SS 316	10–12	140–240 (5)	520–540
PE-16	12–15	160–240	780
Al	15	320–360	300
V-20 Ti	11	59	230
V	12	55–60 (0.3)	105
Nb	7	20–30 (1)	80–105
Mo	7–8	45–50 (2)	95–100
Be	–	3,050	–
C	6	600–3000 (34)	–
B_4C	–	3 600	–
$LiAlO_2$	–	15,500	–
SAP (Al + Al_2O_3)	14	410	780
SiC	–	1,800	580
Al_2O_3	15	435	840

Values in parentheses are helium production rates for a fast fission reactor (EBR-II) [2, 3].

Table 2: Ratio of appm(He) to dpa for a Fast Fission Reactor (FFTF), a Thermal Fission Reactor (HFIR), a Rotating Target Neutron Source of 14 MeV Neutrons (RTNS-II), and a Fusion Reactor Wall [2, 3]

	FFTF	*HFIR*	*RTNS-II*	*Fusion reactor*
Nb	0.033	0.073	5.4	3.3
V	0.004	0.009	9.7	4.9
Mo	0.05	0.012		5.8
Al	0.11	0.31	63	24
316 SS	0.096	95	36	21

Table 3: Solid Transmutation Rates in Fusion Reactor Materials for a Neutron Wall Loading of 1 MW/m^2 [2, 3]

Original metal	*Transmutation product*	*Transmutation rate appm/year*
Al	Mg	400
	Si	40
SS 316	Mn	1200
	V	200
	Ti	50
V	Cr	130
	Ti	80
Nb	Zr	700
Mo	Tc	400
	Ru	30

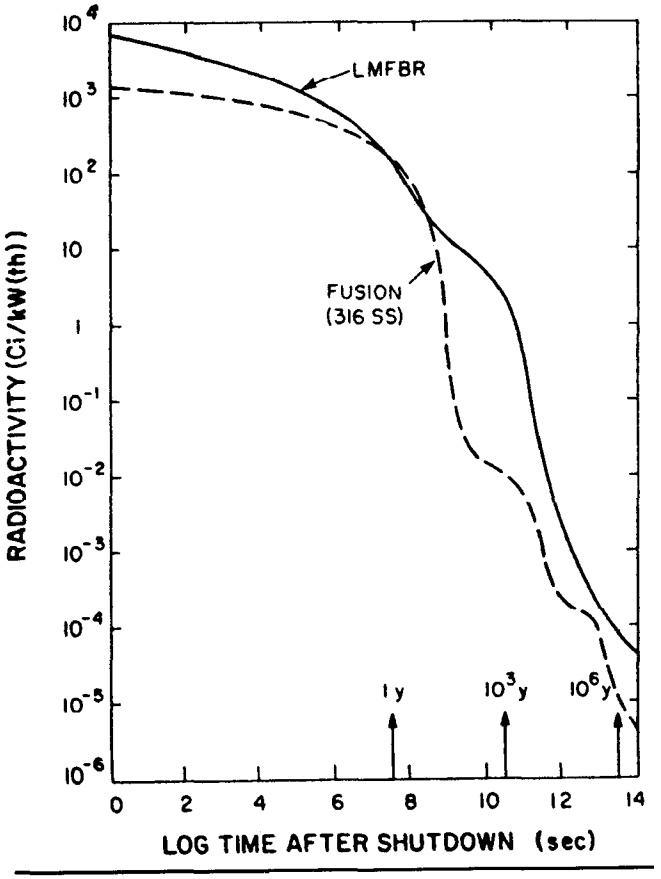


Figure 3a: Comparison of radioactive inventory for fission and fusion reactors with SS 316 structure. From: *Hafele W, Holdren JP, Kessler G, Kulcinski GL* Fusion and fast breeder reactors. Austria: Inter. Inst. Applied Systems Analysis, 1977.

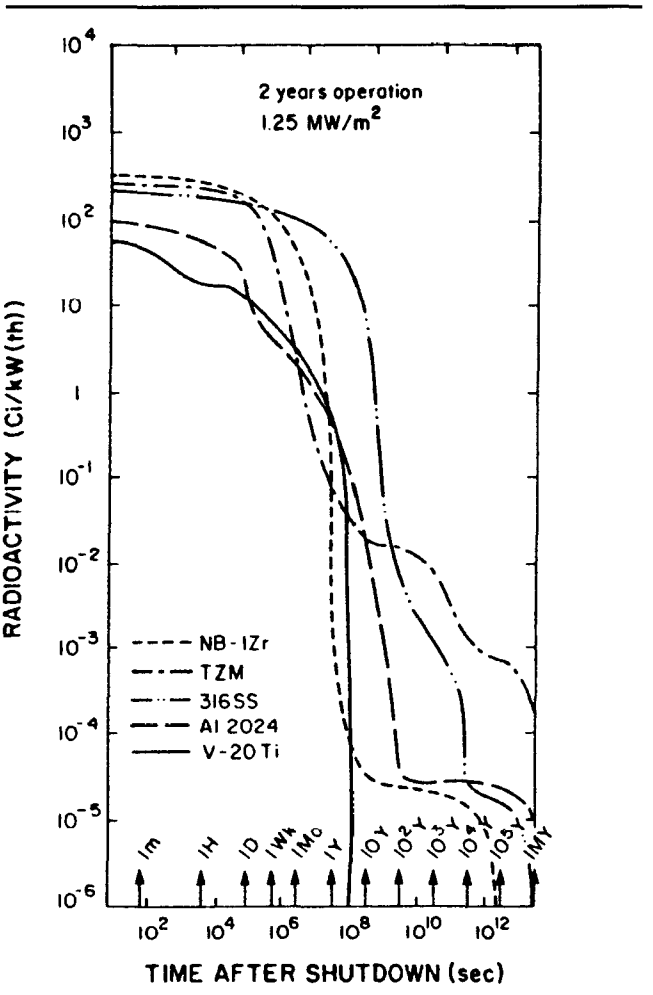


Figure 3b: Radioactivity in fusion reactor first walls after shutdown. As Figure 3a.

ment of intense 14 MeV neutron sources. Without such sources, materials specialists have to rely on test facilities such as fission reactors and ion accelerators, which only partially replicate the fusion neutron environment. In fact it is extremely difficult to achieve a satisfactory fusion neutron source, short of constructing an actual DT, high duty-cycle fusion reactor. The principal requirements of such a materials test source are: 14 MeV neutrons (or at least a very hard spectrum, $\gtrsim$10 MeV) at flux densities $\gtrsim 10^{19}$ n/m^{2}s, with as much test volume as possible > 1 litre. Since a reactor first wall will experience $\sim 10^{18}$ (14 MeV) n/m^{2}s, and since accelerated testing is highly desirable, the test flux density should be at least ten times higher.

In light of these requirements, it is somewhat sobering to consider the currently available and planned neutron sources (Table 4). The RTNS II, currently the most powerful available source, falls short of the above measures by orders of magnitude. The INS (Intense Neutron Source) was initially funded by the U.S. Government ($25 million), but was cancelled in 1977 as part of general economic measures

Table 4: High Energy Neutron Sources for Fusion Materials Studies

Location	*Name*	*Reaction*	*Beam energy (MeV)*	*Target current (mA)*	*Total strength (n/s)*	*Flux intensity ($n/m^2 s$)*	*Test volume (cm^3)*	*Status*
Livermore	RTNSI	d + ^{3}H	0.4	22	6×10^{12}	10^{16}	1	Operated 1970's
Livermore	RTNSII	d + ^{3}H	0.4	150	4×10^{13}	10^{17}	1	Operating
Los Alamos	INS	t + ^{2}H	0.3	1000	10^{15}	10^{18}	1	Cancelled
Hanford	FMIT	d + Li	35	100	10^{16}	10^{19}	10	Cancelled
Needed						10^{19}	$\gtrsim 10^4$	

Table 5: Next Generation of Proposed Fusion Test Facilities [4]

	CIT U.S.	*NET Europe*	*FER Japan*	*INTOR International*
DT fusion power (MW)	300	600	300	600
Pulse length (sec)	4	1000	2000	200
Total pulses	10^4	10^5	10^4	4×10^5
Total burn time[1] (sec)	10^5	10^8	10^7	8×10^7
Total T_2 consumed (kg)	0.05	100	7	94
Fraction of T_2 bred in device	0	0.4	~0	0.6
Breeding material	–	$Li_{17}Pb_{83}$	LiO_2	LiO_2
Time-averaged hard neutron flux[2] ($n/m^2 s$)	2×10^{16}	3×10^{17}	3×10^{16}	3×10^{17}
Total exposure (MW-yr/m^2)	0.02	3	0.3	3

[1]Maximum.
[2]Total exposure averaged over 5 years.

imposed by the Carter Administration. The INS would have employed a d.c. 300 keV 1 Amp T^+ ion beam impacting on a high-speed D_2 flow. A recently considered neutron source is the D-Li stripping FMIT (Fusion Materials Irradiation Test Facility) which the U.S. has proposed to build, at a cost of ~$150 million, provided the cost is shared internationally. The FMIT proposal has been on the table for many years but has not obtained the necessary financial support.

Because of the inherently serious nature of the neutron damage problem, and because the clearly pressing necessity to develop a suitable test facility has not been acted upon, this materials problem may be the most serious one facing fusion reactor development.

The seriousness of this problem is further underlined by considering the next generation of proposed fusion machines [4] (Table 5), the ones which are intended to be the last before an actual Demonstration Reactor, DEMO. These devices will each cost $\gtrsim$ 10^9. Nevertheless, even these impressive facilities will not produce the hard-neutron radiation exposure rates (~10^{19} n/m^2/s), nor even the integrated life-time exposure (~20 MW-yr/m^2) required for fusion reactor development.

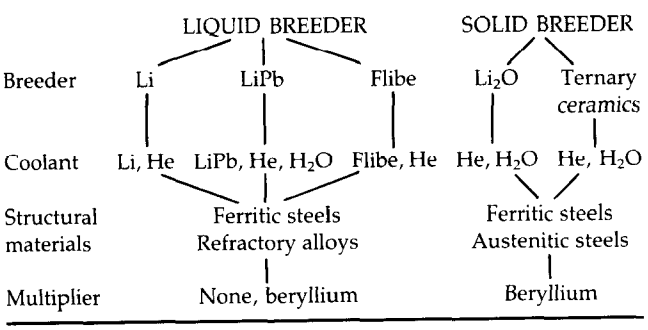
The Tritium Breeding Blanket

Tritium is not found to any significant degree in nature and must be manufactured for DT fusion reactors. The most probable method is to utilize the neutrons produced by the DT reaction itself to bombard a lithium-containing blanket surrounding the reactor, breeding tritium via the reaction ^{6}Li (n, α) T (^{6}Li constitutes ~7.5% of natural lithium, ^{7}Li ~92.5%). The blanket must also incorporate a heat-removal system which is capable of handling the fusion power, 80% of which is deposited directly throughout the bulk of the blanket by the neutrons.

'Blanket issues are highly interrelated to material issues and material uncertainties can strongly affect the feasibility of a blanket concept' [5]. The blanket must contain a wide variety of materials, performing different functions (Table 6), with mutual compatibility [5]:

1. *Breeder material* must contain Li in sufficiently high atomic density to achieve a breeding ratio of at least unity, i.e., one tritium atom created for each 14 MeV neutron entering the blanket. Three categories of breeder material are being considered:

Table 6: Primary Blanket Options [5]



(a) liquid lithium;
(b) solid lithium compounds such as LiO_2, $LiAlO_2$, Li_7Pb_2, and $LiSiO_2$; and
(c) molten lithium salts such as LiF-BeF_2 (FLIBE).

2. *Coolants* which remove the heat and possibly also the tritium. Candidate materials: H_2O, He, liquid Li, liquid Na, molten salt, and fluidized LiO_2.
3. *Neutron multipliers* are likely to be required to achieve a breeding ratio ≥ 1. Candidates: Be, Pb.
4. *Neutron moderators* are likely to be needed to enhance the $^6Li(n, \alpha)$ T reaction, which uses thermal neutrons. Candidates: H_2O, C.
5. *Structural material* for the coolant channels, etc., must be compatible with the foregoing exotic materials. Candidates: austenitic, ferritic, and martensitic steels; Fe–Cr–Ni super alloys; and reactive and refractory metals such as V.

Liquid lithium is, in many respects, the ideal breeding material: it has a high Li density and thus a good breeding ratio (Figure 4). It will therefore probably not require ^6Li isotope enrichment, nor a neutron multiplier; liquid Li is an excellent heat-transfer fluid and can be used simultaneously as coolant; the tritium is naturally removed from the blanket for extraction. Unfortunately a number of serious deficiencies may rule out this simple option:

1. Corrosion of the structural materials by the lithium, combined with the large mass transfer, is likely to result in the plugging of valves, etc., with (radioactive) corrosion products.
2. Lithium reacts readily with O_2, N_2, and H_2O, and the risk of a lithium fire releasing the tritium (in the biologically hazardous oxide form) is possibly unacceptable.
3. Lithium is electrically conducting and a liquid lithium coolant would experience pumping losses due to magnetohydrodynamic effects in circulating through the reactors' magnetic fields.

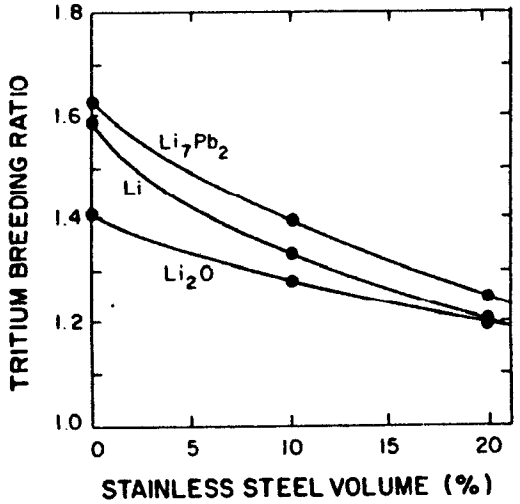


Figure 4: Attainable breeding ratio vs volume per cent stainless steel structure [2].

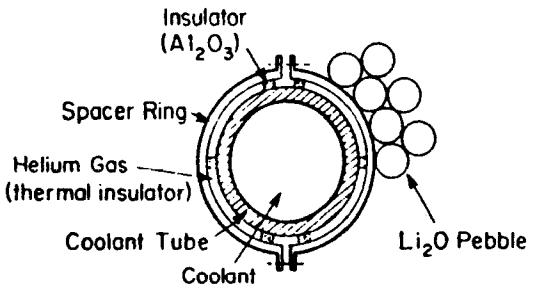


Figure 5: INTOR breeding blanket concept [6].

4. The solubility of tritium in lithium would make it difficult to keep the T-content down to the ppm level required for an acceptable tritium inventory in the blanket: ~1 kg T.

Solid breeders have therefore attracted attention, although this option also encounters materials problems:

1. Tritium self-sufficiency. With lower atomic Li content in solid breeders, the achievement of an adequate breeding ratio is jeopardized. Considering that not all of the periphery is usable for breeding, owing to the presence of structural components (Figure 4), this problem is a serious one. A multiplier is probably necessary, and since the desired neutrons are slow, direct mixing of, e.g., Be and the breeder is desirable. This, however, raises problems of chemical compatibility.
2. Tritium inventory and recovery. The tritium must be rapidly removed from the solid breeder and convected out of the blanket in order to minimize tritium inventory. The transfer of tritium from solid to coolant or purge channel is governed by the largely unknown processes of diffusion, solubility, and surface recombination in materials operating at conditions of elevated temperature, neutron irradiation, and surface chemical contamination.

The INTOR design (International Tokamak Reactor – a conceptional design of the next generation Tokamak experiment by the world's leading fusion countries) [6] calls for a net breeding ratio of >0.6, with a blanket coverage of ~60% and a thickness of ~0.5 m. Since INTOR is an experimental device, not a power reactor, the tritium fuel costs to make up the missing 40% are not prohibitive (although they could still run to ~$1 billion, since an external supply of 4–8 kgm T / year is required, at a cost of $10–100 million / kg). Experimental access limits the blanket coverage, and the bore of the magnets limits the blanket thickness. The latest INTOR blanket option based on LiO_2 is described in Figure 5 and Table 7. The tritium would be carried off in a helium purge circuit, separate from the H_2O coolant circuit.

Plasma Materials Interactions (PMI)

The materials isues associated with the interaction of the plasma with the wall components are the most

Table 7: Parameters for INTOR Reference Li_2O Blanket [6]

Breeder material	Li_2O
Thickness of breeder region (cm)	50
Enrichment of ^6Li (%)	30
Temperature limits for breeder (°C)	410–800
Purge stream for tritium recovery	He (0.1 MPa)
Coolant	H_2O
Coolant temperature (°C)	<280
Structure	316 SS
Maximum structure temperature (°C)	350
Neutron multiplier	Pb
Neutron multiplier thickness (cm)	5
Tritium-breeding ratio	0.65

studied and best understood of the fusion materials problems. This is simply the result of the fact that while the 14 MeV neutron damage and breeding blanket materials questions relate to future machine operation, PMI occurs – indeed dominates – the operating properties of current experimental devices.

Central plasma temperatures, even in current machines, can exceed 10^8 K. Fortunately, the insulating effect of the magnetic field supports a strong temperature gradient across the plasma. Nevertheless, edge plasma temperatures – i.e., of the plasma in actual contact with the walls – are extremely high: one million degrees. Not surprisingly, this results in a strong plasma surface interaction.

PMI leads to erosion of the surface due to a number of processes. Generally, the most serious is *physical sputtering* [7], which is the result of the simple process of momentum transfer from the fast-moving plasma particles to atoms in the solid lattice, knocking them out. The magnitude of this wall erosion process is indicated by the experimentally measured *yield* (Figure 6), which is dependent on the elemental composition

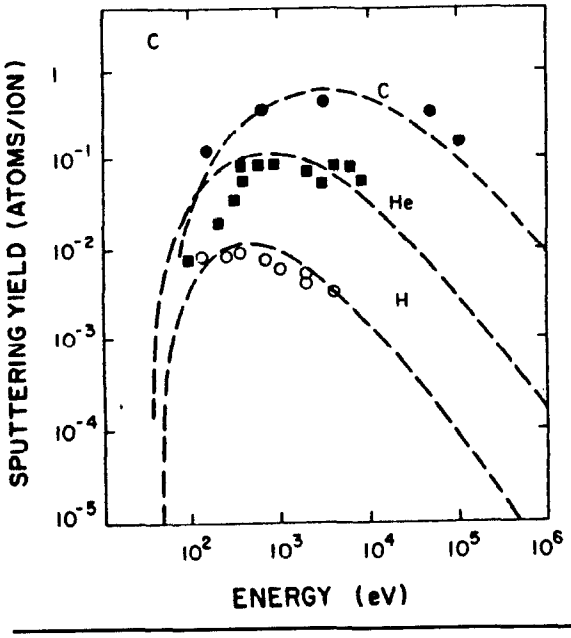


Figure 6a: Sputtering of carbon by H, He, C ions [7].

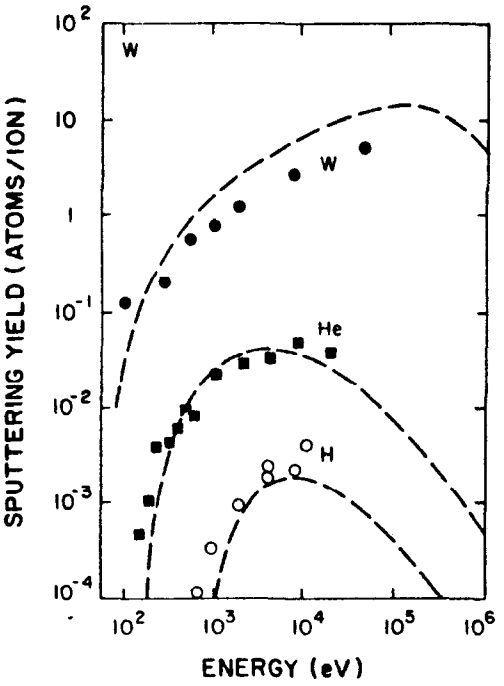


Figure 6b: Sputtering of W by H, He, W ions [7].

of projectile and substrate, and the projectile energy. As indicated, removal rates can exceed ~1 atom removed per ion for impacting energies of a few hundred eV, typical values for an edge plasma of temperature ~10^6 K. Initially, only the hydrogenic ions cause sputtering. However, the sputtered impurity atoms is quickly ionized upon entering the plasma, and in steady-state returns to the solid surface at the same rate, causing *self-sputtering* by its own impact.

Since impurity ions carry more momentum than hydrogenic ones, their sputtering yield is higher (Figure 6). For steady-state conditions, the impurity removal rate Γ_I [atoms/m^2] is the sum of the hydrogenic and self-sputtering rates: $Y_H\Gamma_H$ and $Y_I\Gamma_I$, respectively; Y_H and Y_I are the hydrogenic and impurity yields; and Γ_H is the hydrogenic flux. Therefore,

$$\Gamma_I = Y_H\Gamma_H + Y_I\Gamma_I \quad (1)$$

or

$$\Gamma_I = \frac{Y_H\Gamma_H}{1 - Y_I}. \quad (2)$$

Thus, self-sputtering is not additive but multiplicative; furthermore, as $Y_I \rightarrow 1$, a catastrophic runaway can occur.

High erosion rates are unacceptable for at least three reasons:

1. The wall material, present in the plasma as an unwanted impurity, thermally radiates away the plasma heat content, preventing net energy production.
2. The wall wears out and its frequent replacement is not compatible with economic plant operation.

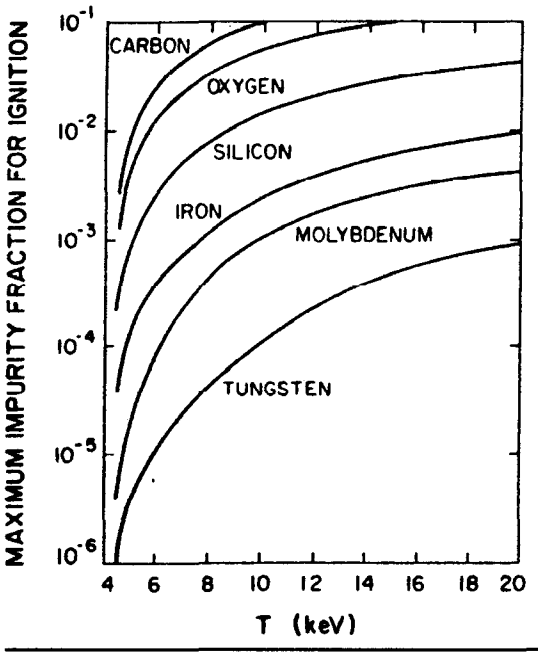


Figure 7: Maximum tolerable impurity fraction for ignition of a DT plasma vs. temperature for various impurity species [8].

3. Gasified impurities such as methane enter the exhaust / clean-up system, which has the task of extracting the highly valuable, unburnt tritium and returning it to the plasma in a completely pure form. Such impurities create one of the biggest problems in the re-cycle loop.

Only the first of these problems is a serious one in currently operating experimental devices. The other two will become more serious as reactor conditions are approached. The plasma contamination problem is perhaps the greatest stumbling block to the demonstration of net fusion power. While the fusion 'flame' burns intensely hot, it is a remarkably vulnerable flame. Were one simply to blow into a magnetic fusion reactor the size of a large room, the fusion flame would be extinguished! (See below.) Eventually, this vulnerability of a fusion plasma will be appreciated as a valuable safety feature: any departure from designed operating conditions of a fusion power reactor will increase the plasma-materials interactions, contaminating and extinguishing the reaction. For the present, however, this vulnerability is an enormous obstacle.

All bodies radiate heat to their surroundings. The hotter the body, the shorter the wavelength of the radiation. Human beings radiate in the infra-red. Fusion plasmas radiate X-rays. This radiation, fortunately, is not black-body, which is so intense that it would result in a hopeless prospect of net energy production at 10^8 K. Rather, the radiation is due to electrons colliding with positively charged nuclei in the plasma. The power of this collisional radiation varies as Z^2 – the charge on the nucleus squared. For this reason, high-Z impurities are particularly damaging (Figure 7). This figure indicates [8] the maximum permitted concentration of various impurities in a DT plasma, which will just permit ignition – i.e., the point at which the self-heating of the plasma by the fusion reaction itself equals the radiative cooling rate. This figure demonstrates how risky it is to employ materials such as tungsten, whose high melting point, good heat-conductivity, and low sputtering rate (Figure 6b) may not be adequate compensations for the low permitted concentration in the plasma (Figure 7). The attraction of low-Z wall materials such as carbon is also evident from Figures 6a and 7. Twenty years ago refractory materials such as tungsten were widely employed in fusion experiments. However, as the seriousness of the impurity problem became clearer, systems were changed to stainless steel and then to carbon. Most operating fusion devices today operate with large quantities of carbon protecting the first walls; indeed, with tons of graphite in the largest machines, such as TFTR at Princeton. Going a step further, beryllium has been experimented with on a small scale in the U.S. machine ISX-B at Oak Ridge [9] and will be tried in the world's largest fusion device, JET, in the next few years. Boron coatings and liquid lithium layers have also been proposed, which, of course, brings one to the end of that line!

With regard to blowing out a fusion flame, let us consider a fusion plasma of typical characteristics: volume of $\sim 100\,m^3$, temperature of $\sim 10^8$ K (corresponding to ~10 keV particle energy), density of $\sim 10^{20}$ D and T per m^3. This yields a total fuel content of $\sim 10^{22}$ D and T ions. With a human breath of about 1 litre of N_2/O_2 ($\sim 2.7 \times 10^{22}$ N_2/O_2 molecules), the plasma contamination would greatly exceed the permitted ~6% level for oxygen (Figure 7).

The direct prevention of ignition by radiative cooling of the *core* plasma, where the fusion reactions occur, is not the most serious problem caused by impurities in current devices. Two other effects are more serious:

1. Fuel dilution. Since the impurities create high-Z ions they fill the plasma with many extraneous electrons for each impurity ion. Each electron adds just as much to the plasma pressure as a D / T fuel ion, and since the confining pressure exerted by the magnetic field, $B^2/2\mu_0$, is limited, the result is fuel dilution. Since the fusions power P_F varies as $n_D n_T$, i.e., n_{fuel}^2, a small impurity fraction reduces P_F enormously, even for low-Z impurities. For example, 5% carbon reduces n_{fuel} by ~30%, hence P_F by ~50%.
2. Density limit. Finite magnetic pressure aside, one would think that n_{fuel} could be raised to any desired level simply by puffing more D_2 or T_2 into the plasma. Unfortunately, an upper density limit occurs for stable operation of the plasma, and at plasma pressures only a small fraction of the available magnetic pressure: ~1%. The cause of this serious limit is not completely understood, but is almost certainly due to impurities, since purer plasmas have higher density limits (Figure 8).

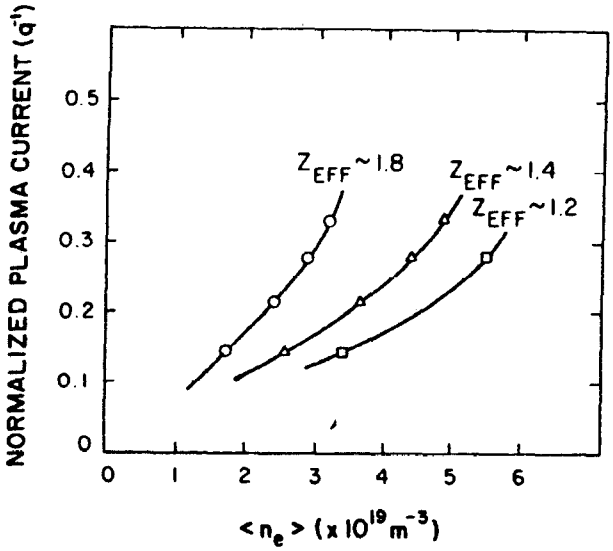


Figure 8: Maximum plasma density achievable at a given plasma current is dependent on impurity level. For pure DT plasma $Z_{eff} = 0$. Calculated result for JT60. *Hirayama T*, et al. J Nucl Mater 1986.

For energy break-even, the fuel density must satisfy the Lawson Criterion:

$$n_{fuel}\,\tau_{confinement} \geq 10^{20}\ [s/m^3], \qquad (3)$$

and thus both fuel dilution and the density limit strike directly at the heart of the entire fusion enterprise.

Because of the critically serious nature of this *near-term* materials problem, fusion research has had to pay greater attention to materials issues than is usually the case for a technology still in its scientific-feasibility phase. The official four objectives of the JET Project [10], for example, given considerable prominence to this area:

scaling of plasma behaviour,
plasma-wall-interactions,
plasma heating,
α-particle behaviour.

The magnitude of the PMI is obviously intimately related to the effectiveness of the magnetic confinement. Perfect confinement would result in no PMI! In reality, of course, the plasma-confining magnetic bottle is quite leaky. Even in the biggest machines, which have the best confinement, the average particle lifetime in the plasma is less than one second. Thus every second, or faster, the entire plasma content of ~10^{22} ions strikes the walls. In principle, this load could be distributed evenly over the entire wall surface ~100 m^2. For various reasons, a more controlled PMI is desirable, and it can be arranged for the magnetic field to channel most of the plasma outflow to special surfaces, called *'limiters'* or *'divertor plates'* (Figure 9). While the latter are only of area $\lesssim$1 m^2, one can afford to make them of special materials and can replace them more readily than the entire wall. The wall still catches

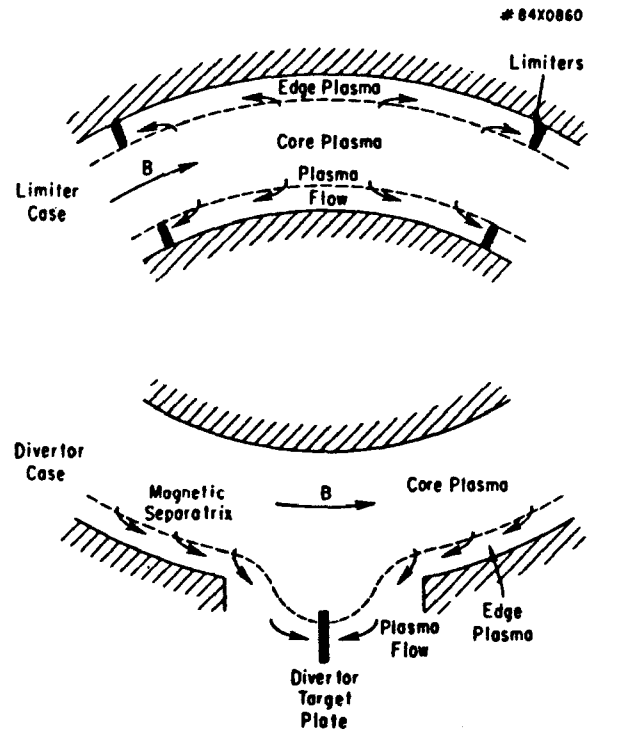


Figure 9: Schematic of limiter and divertor geometry. PMI at divertor target is located further from main plasma than for limiter target.

some action, but orders of magnitude less (per square metre). Thus the PMI occurs at two quite different levels of intensity:

1. Limiters and divertor plates receive 10^{22}–10^{24} ions / m^2s, i.e., ~amps / cm^2.
2. Walls receive 10^{19}–10^{21} ions / m^2s, i.e., ~milliamps / cm^2.

The total incident flux (ions / s) is about the same for the two types of surfaces, and thus the two zones pose about equally important materials problems. The nature of the problems differ, of course. The erosion rate of the limiters, for example, can be quite spectacular. Taking $Y_{D\,on\,C}$ =0.1 for example (Figure 6a), and Γ_D = 10^{23} (D^+ / m^2 s), leads to an erosion rate of 10^{22} C / m^2 s (~0.3 mm / hour), just due to the D^+ impact alone. Since much of this removed carbon may be in the form of methane, the impurity load on the re-cycle loop could be unacceptable (operation of cryogenic isotope separators in the loop requires impurity levels $\lesssim$ 1 ppm). Fortunately, the impurity atoms and molecules upon entering the plasma are ionized and are then caught up in the hydrogenic flow to the surfaces. While this increases the sputtering, it also leads to *re-deposition*. The latter process is clearly of critical importance if acceptable erosion rates and fuel re-cycle loop conditions are to be achieved. The effectiveness of impurities in contaminating the *centre* of the plasma is also greatly affected by the rate at which this

impurity turn-around occurs. The processes involved in this complex pattern are only starting to be understood. As to the materials side of this, it is evident that whatever surface is initially introduced into the device, it will quickly become a re-deposited surface, with its own unique properties. It is therefore important to carry out materials tests on re-deposited materials, and to create these materials in the same way as is actually encountered in a working device.

The foregoing is focussed on only one aspect of PMI – erosion. A number of other important effects also arise:

1. Hydrogen permeation. Hydrogen (including D and T) can permeate through virtually all solids. This results in embrittlement and swelling, and in the case of (radioactive) tritium, in a containment problem. This problem is enormously exacerbated when the hydrogen arrives at the front surface in an already dissociated form (atoms and ions rather than molecules), causing effective permeation rates to increase by many orders of magnitude. Since the fusion wall receives hydrogen in the pre-dissociated form, the problem of T-contamination of the coolant circuit is a serious one.
2. Hydrogen retention. As above, but the problem is now one of keeping the tritium inventory of the plant to an acceptably low level.

In addition, there are all the thermomechanical materials problems associated with cyclical, high heat-loading in a radiation environment to be dealt with.

From the above, it is obvious that PMI problems are major ones. Fortunately, these problems are being addressed and solid progress is being made. As an example, in initial applications [11] of radio frequency wave heating to fusion experiments, even brief bursts ($\lesssim$0.1 sec) of low power ($\lesssim$0.2 MW) caused such a cloud of impurities to come off the RF antennae, that the plasma was actually cooled! Today, the RF antennae on JET inject ~10 MW of power for seconds-long pulses without appreciably changing the impurity level of the plasma [10]. While many PMI challenges lie ahead, particularly as break-even conditions are approached and α-particles contribute to the PMI, etc., a reasonable basis for optimism exists.

Canadian Fusion Materials Research

Canadian research on fusion materials is concentrated on breeder materials and plasma materials interactions. Two projects in these areas – the Chalk River Fusion Blanket Programme and the Tokamak de Varennes – have been extensively reported on and therefore only a brief review is included here. Other fusion materials R & D work underway at the Ontario Hydro Research Laboratory, McMaster University, and the University of Toronto Institute for Aerospace Studies are described.

Breeder Materials R & D at CRNL

A Fusion Blanket Programme was launched during the first five-year programme of the Canadian Fusion Fuels Technology Project (CFFTP), with joint funding by the CFFTP and AECL. The Blanket Programme utilizes a major fission neutron irradiation source at the Chalk River Nuclear Laboratories of AECL (CRNL) – the NRU research reactor. Focus is on irradiation tests of solid breeder ceramics: $LiAlO_2$ (jointly studied with France), $LiAlO_2$ (with Japan), and Li_2O (with the U.K.). Both vented (CRITIC Project) and unvented (CREATE Project) capsule irradiation tests are performed to establish the release of tritium from the ceramic and to evaluate breeder / cladding interactions, swelling, cracking, etc. The (fission) neutron damage reproduces the displacement rate of a fusion reactor near the blanket rear, but is short by two orders of magnitude for the front (first wall).

With regard to the fabrication of breeder ceramics, the CRNL Programme is focussed on the fabrication of $LiAlO_2$ microspheres by various novel techniques, such as an organic sol-gel process.

Lithium (^6Li) isotope enrichment techniques are under study with the purpose of evolving new, environmentally attractive approaches.

Liquid breeder / coolant options are under investigation, including liquid metal breeders, and organic coolants, the latter based on the twenty-year operating experience of AECL's WR-1 organic cooled heavy-water-moderated reactor.

A major alternative to the conventional blanket approach for producing tritium is a new concept based on a water-cooled blanket employing lithium salts dissolved in light or heavy water. This novel approach is the object of a joint study by CFFTP in collaboration with Grumman and Rensselaer Polytechnical Institute.

Materials Studies on the Tokamak de Varennes

The scientific programme of the Tokamak de Varennes is centred around the study of impurity transport and control, plasma-wall interactions, the effect of long pulses, and materials studies. The device will have a divertor system (Figure 9), high-speed pumping of the vacuum vessel and advanced diagnostics – all intended for a strong programme in PMI. The work will include studies of:

1. the equilibrium (long pulses) between the edge plasma and the wall;
2. the thermal fatigue of the materials exposed to large heat deposition (combination of long pulses and a tight, high-energy flux divertor configuration);
3. sputtering, redeposition, and net erosion;
4. coatings, such as titanium-carbide-coated carbon tiles;
5. divertor operation and impurity accumulation;
6. recycling of the hydrogenic species in long pulse discharges, where plasma-solid equilibrium is approached; and

7. the characterization of materials for the limiters and divertor plates.

The Fusion Materials Group at the Institut National de la Récherche Scientifique (INRS) is pursuing two supporting areas of activity:

1. Development of analysis techniques. A 400 keV accelerator is used for the nuclear micro-analysis of H and D in samples. Detection at the ~1% atomic (H) level, with depth resolution to 100Å, has been demonstrated; laser desorption mechanisms are studied in order to make this into a quantitative tool.
2. Materials development and characterization. This work is carried out in collaboration with the Industrial Materials Research Institute at Boucherville. Thick (300 μm) plasma-sprayed coatings have been demonstrated to have valuable properties: thermal shock resistance if deposited under argon atmosphere; absence of hydride formation or blisters under hydrogen implantation; acceptable porosity and gas content.

Fusion Materials Studies at OHRD

Fusion materials work at the Ontario Hydro Research Division (OHRD), part of the CFFTP Project, focuses on the development and study of materials to be used in ancillary systems, such as tritium storage and purification systems, a secondary containment detritiation system, and a reactor exhaust treatment system. The areas of activity are therefore development of tritium gas handling techniques, hydrogen permeation, and hydrogen gas interaction with hydride formers.

Hydriding studies of uranium and titanium sponge have led to the development of 5 KCi and 500 KCi storage beds for tritium.

Studies on the removal of free or chemically bound tritium from process streams, using zirconium alloys, demonstrated that concentrations as high as 7 Ci / m^3 could be removed from inert gases, indicating that routine tritium scavenging from inert gas process streams is viable.

In conjunction with the Jülich Textor Group (Federal Republic of Germany), field work studies on the Textor Tokamak have led to the development of a permeation probe to monitor atomic hydrogen fluxes to the wall. Related to this, the performance of a hydrogen permeation pump based on composite membranes is under investigation.

A tritium test facility at the 1 KCi level is being commissioned at OHRD for fusion and fission studies.

Fusion Materials Studies at McMaster University

Fusion materials research at McMaster University is part of the CFFTP and focusses on tritium in materials. A project is also underway on neutron damage.

Special coatings are under development to act as tritium permeation barriers, specifically SiC and Al_2O_3 deposited on Ni substrates via various techniques, such as Al evaporation followed by anodization. The diffusivity and solubility of tritium in the materials can be monitored continuously by measuring the T content using a high-energy tandem accelerator and nuclear reaction analysis (NRA).

The diffusion of D and T in materials such as Ni is studied subject to temperature gradients and radiation damage. The T is ion-implanted in thin foils, sandwiched between permeation barriers of Al_2O_3. The diffusion is thus constrained to be lateral and is monitored by NRA as a function of time and sample temperature.

A neutron damage project is being carried out in collaboration with the U.S. Sandia Laboratory, who load 316 SS samples with high quantities of T at 20,000 psi, 300°C. The T decays to ^{3}He, creating levels of up to 500 appm He. The resulting He bubble formation results in the same fatique and creep problems associated with 14 MeV neutron exposure. The samples are fatigue-tested at McMaster under monotonic and cyclic loading.

Plasma Materials Interaction Studies at the University of Toronto

The fusion materials research at the University of Toronto Institute for Aerospace Studies (UTIAS) is focussed on PMI and includes both laboratory testing and field work on fusion devices outside Toronto. The UTIAS work is part of the CFFTP.

World-wide, perhaps half of all PMI research is carried out using relatively small-scale test facilities, such as ion accelerators, to simulate PMI – but in highly controlled conditions. The other part of PMI research is carried out directly on large experimental fusion devices principally tokamaks. PMI research on working devices has the obvious advantages of focus and relevance, but suffers from complexity, experimental inflexibility, and difficulties in interpreting results, since many processes occur simultaneously. The optimal approach, therefore, appears to be to combine these two avenues of research.

Controlled laboratory studies of plasma materials interactions performed at UTIAS include the following:

1. Chemical erosion of graphite and other first wall materials (e.g., amorphous hydrogenated carbon films produced in the TEXTOR tokamak) using a high-current, low-voltage hydrogenic ion accelerator. The released hydrocarbons, principally methane, are detected mass-spectroscopically. Simultaneously, the substrate can be bombarded by known, controlled fluxes of energetic ions, neutral hydrogen atoms, and electrons to investigate *synergistic* erosion – reproducing the combined exposures to which actual working surfaces are subject. The edge structures in fusion devices are bombarded by about equal fluxes of energetic ions H^+ (100's eV energy) H^0 atoms (~1 eV), and electrons (~20–100 eV). By itself, the H^0 is much less

reactive than the H^+; however, a strong synergistic effect has been discovered for combined H^+ and H^0 exposures. The addition of electrons to the bombarding species does not significantly affect the reactivity of graphite. The UTIAS synergistic studies are carried out collaboratively with the Institute for Chemistry Group, Jülich, Germany.

2. Current research on graphite erosion also includes angle-of-incidence dependence of physical sputtering due to low energy (20–100's eV) ions (H^+, D^+, He^+), and radiation-enhanced sublimation at elevated temperatures ($1{,}200 < T < 2{,}200$ K).
3. Hydrogen permeation is studied using a permeation membrane facility with *in situ* surface analysis of contaminants by Auger Spectroscopy. The level of surface impurities strongly influences the permeation and recombinative-release (fuel re-cycle) of hydrogenic species. A small change in carbon impurity levels on the surface of Pd, from ~0.4 to ~0 monolayer, is found to increase the rate of hydrogenic recombinative-release by more than two orders of magnitude. Since all metal surfaces in a working device become contaminated with carbon, it is clearly important to employ materials data for the relevant, i.e., contaminated, state. Permeation studies with bilayer materials for potential use as a unidirectional hydrogen pump are carried out collaboratively with OHRD and the Jülich Textor Group.
4. Hydrogen inventory effects are studied using thermal desorption and laser-induced desorption of wall samples exposed to pre-dissociated hydrogenic species. Such studies on carbon, carried out jointly with the U.S. Sandia Laboratory (Albuquerque), revealed that the highly porous nature of carbon provides additional pathways along porous internal surfaces – beyond normal diffusion – for hydrogen penetration and retention. Both JET and TFTR have recently experienced unusual plasma-pumping effects, which may be related to carbon porosity.
5. A low-level tritium laboratory (~1 Ci) has just been commissioned in which impurity production, carbon erosion, hydrogen permeation, and inventory effects will be studied using the isotope of interest – tritium – rather than simulating it with protium or deuterium, as hitherto.

Field work studies are carried out on U.S. devices – principally TFTR at Princeton – and European devices, JET and the UKAEA's DITE tokamak, both at Culham, Oxfordshire. Studies include the following:

1. Chemical erosion of graphite under actual operating conditions. A special highly instrumented graphite limiter inserted into DITE is examined spectroscopically as its temperature is varied, in order to test for chemical erosion (which is known to have a strong temperature dependence from accelerator studies). The inferred chemical erosion rates for the DITE experiment were found to be low, in agreement with laboratory beam simulation studies, which indicate reduced yields at high incident current densities, low energies, and in the presence of metal surface contaminants.
2. An inverted-geometry limiter installed in DITE is being tested for its capability of reduced impurity contamination of the plasma. The shape of conventional limiters tends to project sputtered atoms toward the plasma; inversion projects atoms toward the wall. It remains to be demonstrated, however, that inversion is consistent with the other PMI constraints (see 4 below).
3. Edge plasma measurements in JET and TFTR using Langmuir probes indicate that PMI conditions differ in these large machines from earlier ones, in that higher plasma temperatures prevail; i.e. $\sim 10^6$ K rather than the $\sim 10^5$ K more typically encountered in smaller tokamaks. Due to the greater distances between limiter surfaces on these big machines, there is more opportunity for various PMI-related effects to arise than in smaller devices; e.g., radiative cooling of the edge plasma, local ionization re-cycle of the hydrogenic fuel, etc. In addition, the re-deposition patterns are clearer since the long pulses lead to greatly increased total exposure times. With individual discharges on TFTR and JET lasting up to 20 seconds, the total edge data accumulation exceeds that from all previous work on other fusion devices.
4. A Monte Carlo computer code has been developed to track the fate of impurity atoms as they are sputtered from the limiter, enter the edge plasma and ionize, wander off into the central plasma radiating energy, and eventually return to the limiter to sputter and re-deposit. Various limiter geometries, such as the inverted one above, were tested out in a search for optimal designs subject to the constraints of a) distributing the heat load, b) minimizing the impurities reaching the main plasma, c) minimizing net erosion via homogeneous distribution of re-deposited ions, and d) minimizing the build-up of trapped tritium by the process of co-deposition (impurity burial of tritium).

   The co-deposition process was only recently identified, on JET, as possibly the major mechanism governing the tritium inventory tied up in the limiters; each carbon atom deposited buries ~0.5 of a hydrogen atom. The co-deposition and re-deposition processes are intimately related, and since all surfaces in working devices, regardless of their installed materials properties, end up in the re-deposited state, it is essential to understand fully and to control these materials-modification processes.

Conclusions

The achievement of an environmentally and economically attractive energy option based on the fusion process will depend on the solution of a formidable array of materials problems. Three of the most critical relate to 14 MeV neutron damage, tritium breeding, and plasma-surface interactions. Canada has now become a significant participant in the world fusion effort and has undertaken substantial initiatives in two of the material areas: tritium breeding and plasma-surface interactions.

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