Key Materials Issues for Near Term Fusion Reactors

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Abstract

Materials problems in near term fusion devices tend not to be dominated by structural alloys, but more by specialized components. Prevention of radiation damage to superconducting magnet materials (insulators, stabilizers and superconductors) tends to have a large impact on the size, and hence the cost of fusion devices. Other problems such as impurity control components, breeder materials, and coatings for ICF laser optics will also be the focus of near term materials research and sometimes may require solutions which are not commercial reactor relevant.

1. Introduction

As materials scientists participate in the design of the next round of fusion devices (CIT [1], NET [2], FER [3], INTOR [4], and SIRIUS-M [5]), it is clear that for the first time, the question of radiation damage by neutrons will become a significant consideration. In the current magnetic confinement devices such as JET [6] and TFTR [7] the 14 MeV neutron fluence is expected to be so low ($< 10^{16}$ n cm^{-2}) that radiation effects will be inconsequential and only the induced radioactivity will be a problem. Similarly, the 14 MeV neutron fluences in Inertial Confinement Fusion (ICF) devices such as PBFA-II [8] are expected to be less than 10^{15} n/cm^2 and those in the projected Target Development Facility [9] are on the order of 10^{17} n cm^{-2}.

The current designs for the Compact Ignition Tokamak (CIT) [1] in the United States, NET [2] in Europe, FER [3] in Japan and INTOR [4] at the IAEA call for 100's of MW's of DT power to be produced and (except for the CIT) total 14 MeV neutron exposures of $> 10^{21}$ n cm^{-2} to be accumulated. Furthermore, the production of such large fusion powers will require a significant amount of the T_2 which is consumed to be bred in the device itself. A total T_2 consumption of at least 90 kg is envisioned for INTOR alone.

In contrast to commercial power plant operation, the main materials problems for the next round of fusion devices will not be the survival of the structural members of the first wall on the blanket. Instead, the critical problems will be associated with the successful operation of the impurity control schemes, the operation of the superconducting magnets or reflective optics in an irradiation environment and, when used, the successful performance of solid breeder compounds. The objective of this paper is to review what is required for a few selected materials in these components and to identify some of critical data that is needed for their successful operation.

2. Description of Fusion Devices Expected to Produce Significant Neutron Fluxes to Structural Alloys

Table 1 summarizes some of the key parameters for fusion neutron producing devices which might be built in the 1990's or early 21st century. For comparison, the Compact Ignition Tokamak [1] (CIT) is included to illustrate the magnitude of neutron exposure expected in a copper coil, tokamak physics device. The SIRIUS-M design [5], a symmetrically illuminated laser target and materials test facility, is also included as an example of what one might expect from the ICF approach even though such a facility is far less well detailed than the NET/FER/INTOR designs.

It can be seen that in addition to 100's of MW of fusion power, the neutron wall loadings vary from 0.7 to 7 MW/m^2. The burn time for the CIT device is only a few seconds, but all of the other tokamak designs envision 200-2000 second burn periods. (The burn time for an ICF device cannot be compared on the same basis as the neutrons are incident on the wall in nanosecond bursts but take microseconds to completely slow down.) Total burn times are in the 0.003 to 3 FPY (Full Power Years) range meaning that the first walls are exposed to 0.02-5 $MW\text{-}y/m^2$ over the anticipated life of the devices.

Lifetime tritium consumption in these devices ranges from 0.05 to 100 kg and only NET and INTOR plan to use a breeding blanket to provide some of the tritium required for the operation of the devices. The FER and SIRIUS-M devices would have to buy most of the tritium consumed (7-37 kg) although some T_2 could come from test breeding blanket modules.

Table 1.

Summary of Current Design Parameters for the Next Step Fusion Devices in the World Fusion Program

	Magnetic Confinement				Inertial Confinement
	Device				
	CIT[a]	NET[b]	FER[c]	INTOR[d]	SIRIUS-M[e]
	Early 90's	Late 90's	Late 90's	Late 90's	~ 2000
DT Fusion Power - MW	300	600	297	570	134
First Wall Neutron Loading MW/m^2	7	1	0.68	1.3	2 (time ave.)
Pulse Length - s	3.7	200 - 1000	2000	200	$\sim 10^{-6}$
Total Cycles	3×10^3 - 5×10^4[f]	10^5	1.1×10^4	4×10^5	1.6×10^9
Total Burn Time - s	1×10^5	$2\text{-}10 \times 10^7$	1.1×10^7	8×10^7	NAppl.
Total FW Exposure $MW\text{-}y/m^2$	0.02	0.63 - 3.15	0.3	3.3	10
Total T_2 Consumed kg	0.05	26-100	7	94	37
Fraction of T_2 Bred in Device	None	0.3-0.4	Test Mod. Only	0.6	Test Mod. Only
Breeding Mat.	None	$Li_{17}Pb_{83}$ or Li ceramic	Li_2O	Li_2O	None
Max. Neut. Fluence TF Coil n/cm^2 (E > 0.1 MeV)	1×10^{17}[g]	5×10^{17}	1.2×10^{17}	3×10^{17}	8×10^{20} (mirror)

(a) Reference 1
(b) Reference 2 - Double Null
(c) Reference 3
(d) Reference 4
(e) Reference 5
(f) 3,000 full pulses, 50,000 partial pulses
(g) normal copper coil

Finally, the radiation exposure to the superconducting magnets is calculated to be 1 to 5 x 10^{17} n/cm^2 (E > 0.1 MeV) over the life of NET, FER, and INTOR. The 1-5 x 10^{17} n/cm^{-2} exposure includes neutrons of all energies greater than 0.1 MeV leaking from the magnet shield and, as we will show later, represents a very conservative design point for NbTi superconducting magnets.

The rest of this paper will focus on 4 issues which are crucial to the economics and successful operation of these devices: radiation effects to superconductors, plasma interactive components, "cold" tritium breeding and pulsed damage in ICF devices.

3. Radiation Damage to Superconducting Magnets

3.1 Damage Limits and Cost Impact

There are generally 5 accepted design criteria for the successful operation of superconducting magnets in a radiation field:

1. Thermal Insulation
   The operating limit is usually set by the degradation of mechanical strength or ductility of the thin films (usually mylar or Al coated glass paper).
2. Electrical Insulation
   Electrical breakdown, compressive, flexural, or tensile strength determines useful life.
3. Stabilizer Material
   Irradiation induced electrical resistivity coupled with cryogenic stability criteria dictates the amount of stabilizer needed to protect magnets.
4. Superconducting Filaments
   Degradation of T_c, J_c, or H_c properties with irradiation determines useful life.
5. Nuclear Heating
   The deposition of heat in the magnets presents a load to the cryoplant that must be economically removed.

The design limits for radiation exposure to superconducting magnets have been discussed for over 15 years and Table 2 summarizes one recommended set from the "Workshop on Radiation Limits to Superconducting Magnets" held in

Table 2.

Current Design Limits for Superconducting Magnets [10,13]

Area of Concern	Example	Design Limit
Thermal Insulation	Al sheets with glass paper	"No Practical Limit"
Electrical Insulation	Spaulrad-S-polyimide (Compression only)	10^{12} rad
Stabilizer	Enough copper to operate with $\Delta\rho_r$ = 300 nΩ-cm	"No Practical Limit"
Nuclear Heating	Winding Pack	Economics (~ few mW/cm^3)
Superconductor	NbTi	J_o Sat. at 80%, > 10^{20} n/cm^2 (E > 0.1 MeV)
	Nb_3Sn	10^{19} n/cm^2 Fusion Spectrum (E > 0.1 MeV)

Madison, WI, May 23, 1985 [10]. It is not the intent of this paper to discuss the details of these limits as they are discussed elsewhere [11,12]. However it is worthwhile noting that over the past decade the fusion community has been able to raise the allowable exposure of the magnets in some cases by factors of 10 or more, thereby reducing the shielding and the reactor cost. It is also worthwhile noting that improved data and design solutions have been found to remove practical radiation limits to the exposure of thermal insulation, stabilizers and NbTi filaments [11]. The removal of nuclear heating is an economics consideration which can be less restrictive than the limiting design criteria for electrical insulator exposure and neutron effects on Nb_3Sn. Sawan [13] has shown that there is a "rule of thumb" relationship that holds to within a factor of 2 between the various exposure parameters which is stated below for a full 30 full power year (FPY) exposure to the nuclear radiation leaking from the back of a reactor shield:

$$5 \times 10^{10} \text{ rad} \cong 5 \times 10^{19} \text{ n/cm}^2 \; (> 0.1 \text{ MeV}) = 10^{-3} \text{ dpa/FPY} \cong 1 \text{ mW/cc} .$$

From Table 2 and the above relationship we see that the problem first faced in high field magnets using Nb_3Sn will be the loss of critical current density. At that point the values of the other damage parameters are far below the design limits for insulators, stabilizers and nuclear heating. This is depicted in Fig. 1. Obviously if we could develop a Nb_3Sn superconductor which will withstand 5×10^{19} n/cm^2 (E > 0.1 MeV) fusion spectra neutrons, then one would have to begin to worry about the cost of removing the nuclear heat in the magnets. If that could be achieved 10 times more economically, then the development of more radiation damage resistant electrical insulators would be needed.

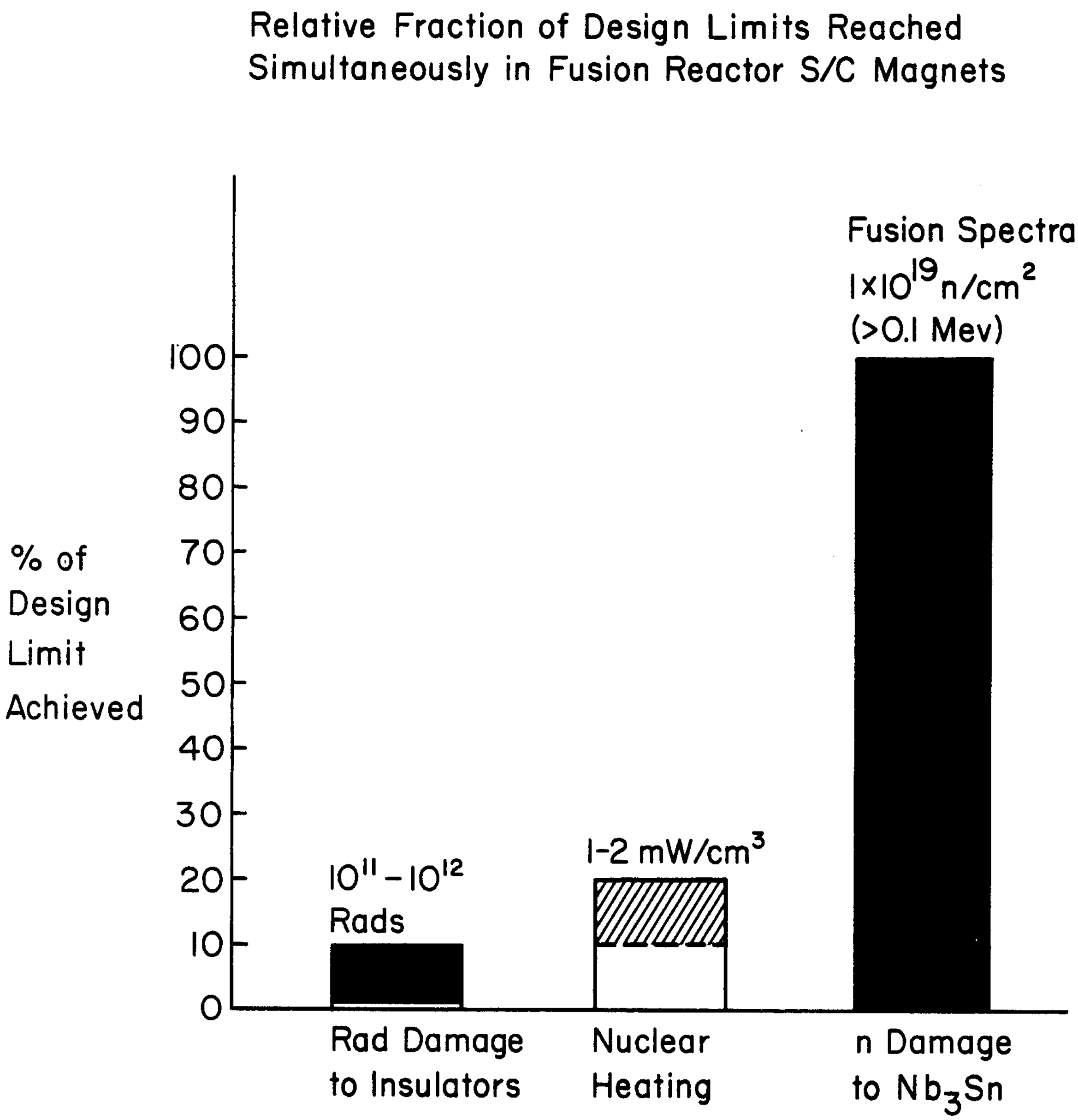


Fig. 1. Relative fraction of design limits reached simultaneously in fusion reactor S/C Magnets.

The magnitude of the economic impact which can be made by developing more radiation damage resistant magnet materials can be seen by considering 3 recent reactor designs: TFCX [14], STARFIRE [15], and MARS [16]. Noting the effectiveness of neutron shields on radiation exposure to magnets, El-Guebaly has shown how the critical damage parameters depend on shield thickness [17]. From Fig. 2 we can see that each cm of shield reduces the radiation exposure by 14-17% for the critical parameters of interest. In other words, a shield of 12-15 cm can reduce the damage in magnets by a factor of 10 from that produced by the neutron and gamma spectrum emerging from a breeding blanket such as that used in the MARS design.

The cost impact of adding or subtracting a cm of shield to the MARS design is estimated to be 2.5 million dollars in direct capital costs and Schmidt [18] has estimated the same number to be 3 M$/cm in TFCX. For STARFIRE, the value is more like 3-4 million dollars per cm of shield thickness removed. Based on these numbers one can see that a factor of 2 improvement in radiation damage resistance for Nb_3Sn could amount to a savings of 9 to 14 million dollars per reactor. A factor of 10 improvement in radiation damage resistance could save 30-40 million dollars on the direct costs per reactor! Such a high return on research investments should be valuable information in allocating source financial resources in materials programs.

3.2 Recent Developments in Damage Correlation

A very important point was recently made by Guinan and co-workers [19-21] concerning the comparison of radiation damage in fusion and fission facilities. They have found that the degradation of superconducting properties is proportional to the damage energy deposited in the material, not simply the total neutron fluence. The damage energy is defined as:

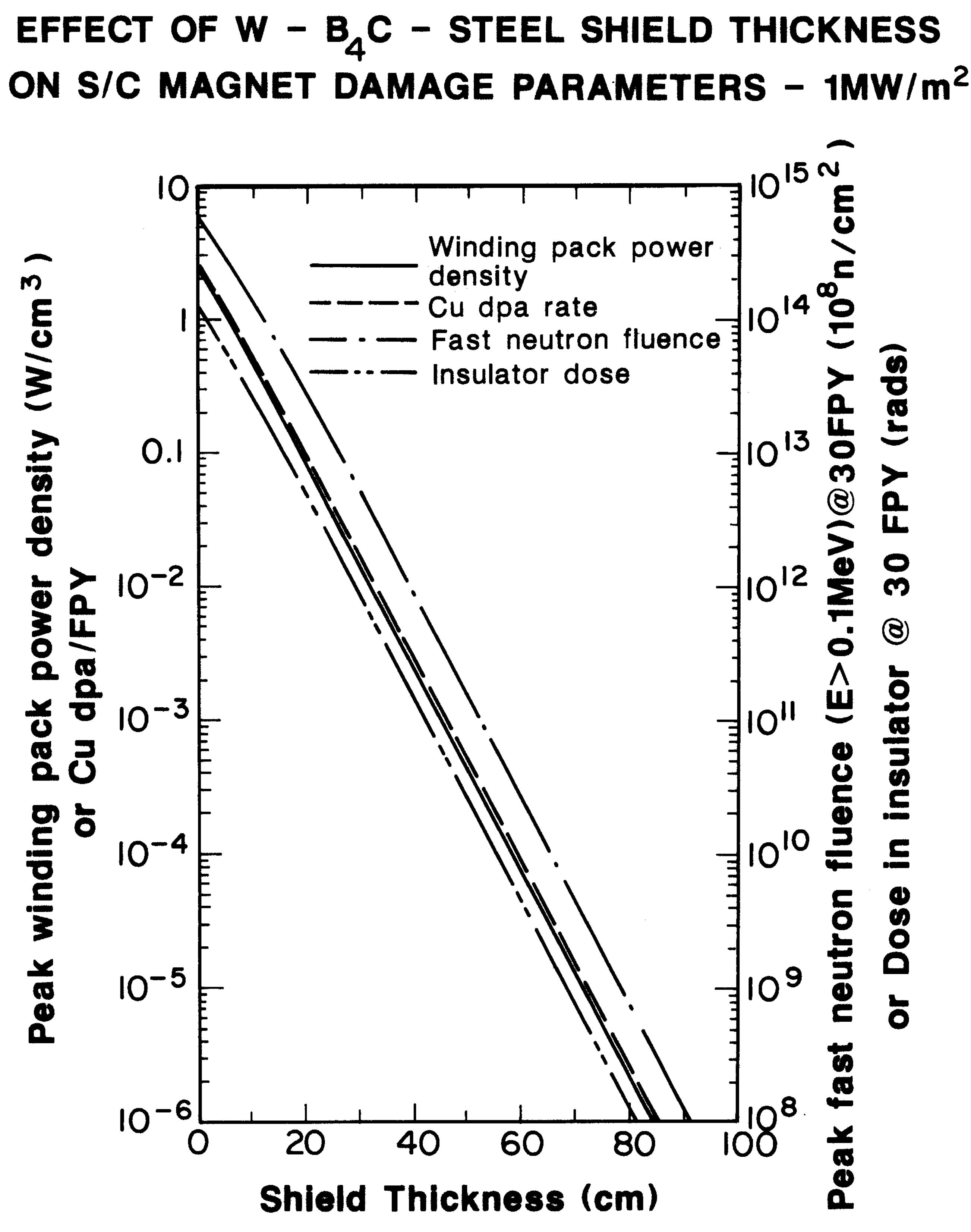


Fig. 2. Effect of W-B_4C/steel shield thickness on S/C magnet damage parameters - 1 MW/m^2 first wall loading.

$$\langle \sigma \cdot T \rangle \, \Phi = \left[\frac{\int \sigma(E) \cdot T(E) \cdot \frac{d\Phi}{dE} \, dE}{\int \frac{d\Phi}{dE} \, dE} \right] \Phi$$

where $\sigma(E)$ = differential neutron scattering cross section

T = primary recoil spectrum

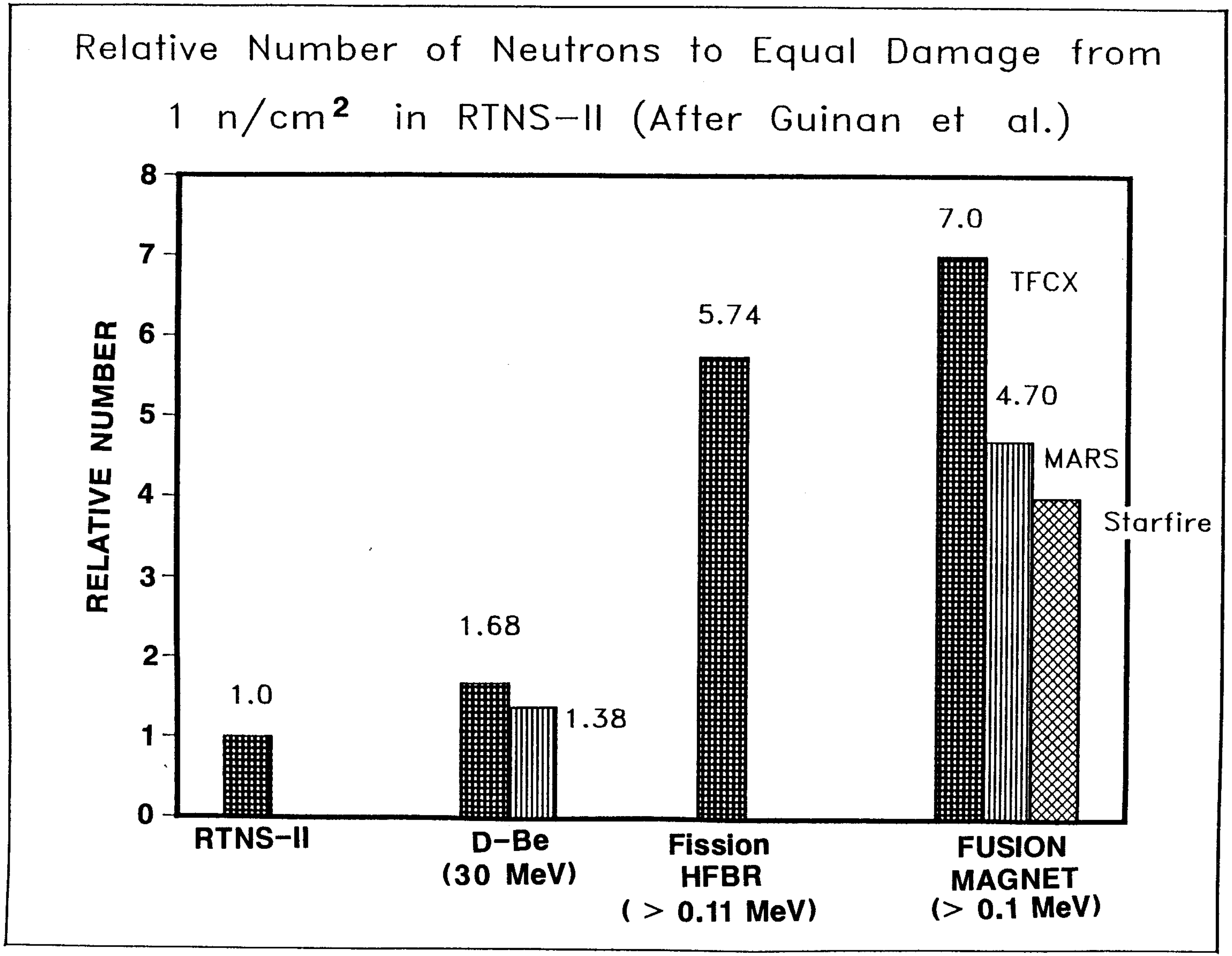
Φ = neutron flux

E = energy.

Such an analysis allows a comparison of the damage obtained from different neutron spectra to be presented on the same scale (avoiding the conversion to dpa which is model dependent) and can provide quantitative support for design limits in Table 2. For example, using this approach, Guinan et al. [20] found that one can characterize the "true" damage potential of a given neutron spectrum as shown in Fig. 3. In this figure the damage energy characteristic of various neutron facilities has been converted into neutrons required to produce the same "damage" as 1 n/cm^2 from RTNS. For example, it takes from 1.38 to 1.68 n/cm^2 from D-Be sources to produce the same effect as 1 DT neutron. Similarly it takes 5.74 n/cm^2 ($E > 0.1$ MeV) from a fission reactor such as HFBR or as many as 7 neutrons ($E > 0.1$ MeV) streaming from the back of the TFCX shield to duplicate 1 DT n/cm^2. The STARFIRE [15] and MARS [16] reactors have different equivalences due to blanket and shield dissimilarities.

Figure 4 is a good example of how this equivalency works for the change in critical current density of Nb_3Sn. The critical current density of monofilament Nb_3Sn irradiated at 4.2° K with 14.5 MeV neutron continues to improve up to ~ 1.5 x 10^{18} n/cm^2 [20]. When this fluence is converted to an equivalent TFCX spectrum it is found that this rise continues to ~ 10^{19} n/cm^2

Fig. 3. Relative number of neutrons which equal damage in niobium superconductors in RTNS-II; after Guinan and coworkers [19-21].



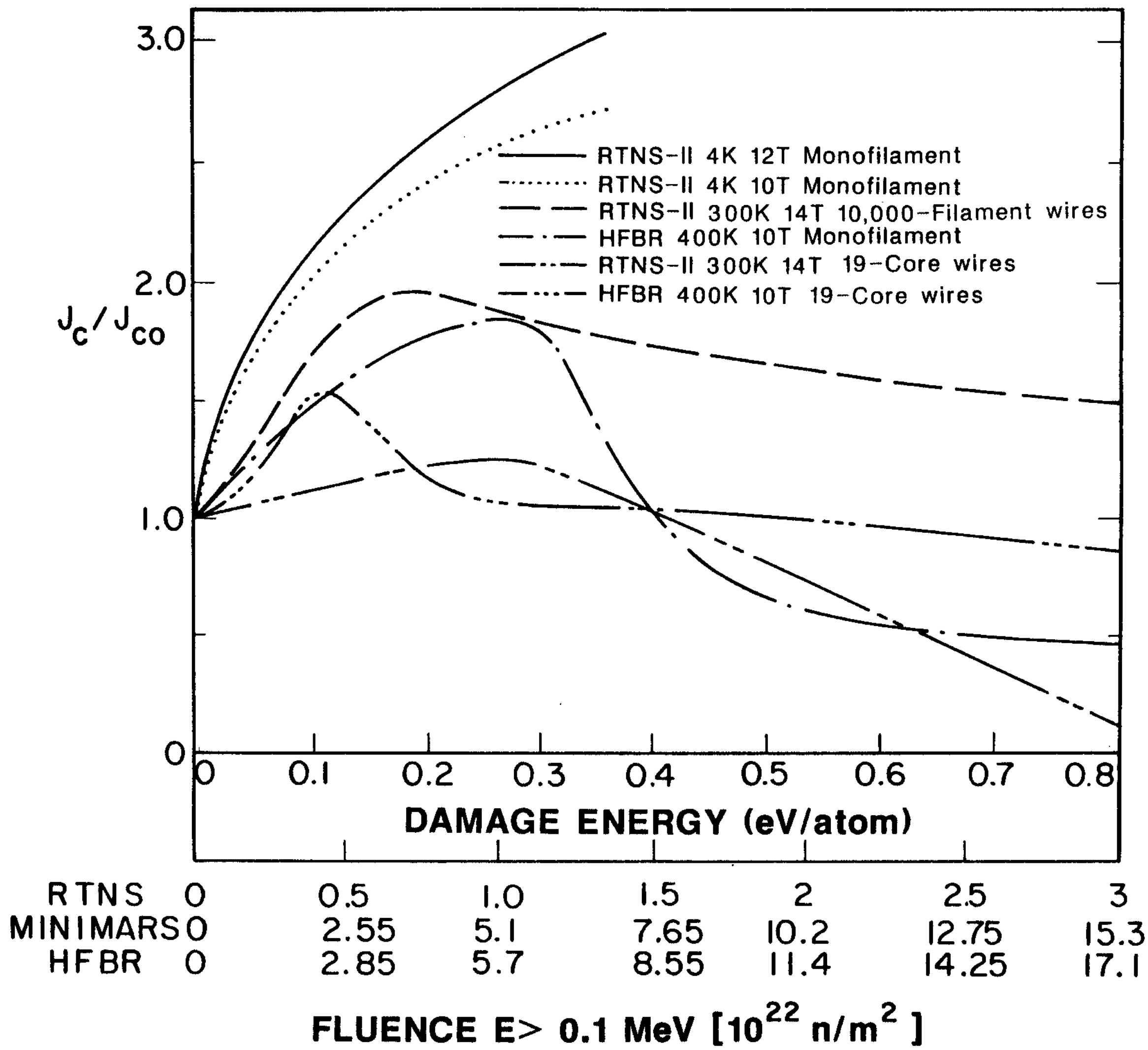


Fig. 4. Critical current changes in Nb_3Sn superconductors as a funtion of neutron fluence [19-22].

or 5 x 10^{18} n/cm^2 (E > 0.1 MeV) in a fission spectrum like HFBR. In addition to the monofilament Nb_3Sn irradiated at 4°K, similar results are obtained from monofilament wire in HFBR at RT [21] and 19 core samples irradiated at RT in RTNS-II [22]. Such conversions will be extremely valuable in the future until high intensity neutron sources become available.

4. Plasma Interactive Components - Near Term Issues

The term Plasma Interactive Components (PIC) is meant to cover any component inside the vacuum vessel which is in contact with the plasma and is usually limited to magnetic devices exclusively. Such PIC's include pumped limiters, divertor plates, RF antennae, neutral beam dumps, as well as the entire inner surface of the vacuum vessel. Each of these components will experience very high heat fluxes and neutron fluxes simultaneously and they will be bombarded with hydrogen isotopes, such as tritium, which could diffuse into the coolant thus contributing to contamination and potential safety problems.

The anticipated operating conditions for these PIC's are listed in Table 3. These components are expected to operate at heat fluxes up to ~ 5 kW/cm^2 for a few seconds in present devices, up to 2000 seconds in near term facilities, or at steady state in commercial units. At the same time these components must withstand heat fluxes of up to 500 kW/cm^2 from disruptions for fractions of milliseconds in present devices or up to 100 ms in near term and commercial units. At the present time, the surface materials subjected to these conditions include TiC coated C, graphite, Be, or Mo alloys. These materials are bonded to structural steels, Ni alloys or Mo alloys and cooled with water. However, none of these materials will be subjected to significant neutron bombardment in present devices, and T_2 permeation is not considered a problem.

Table 3.

Expected Operating Requirements for Plasma Interactive Components

Parameter	Present Devices	Near Term	Commercial Reactors
Peak Heat Flux - kW/cm^2			
Normal Operation	0.4-5	0.5-1	0.5-1
Disruptions	80-500	500	500
Pulse Length			
Normal - s	1-5	5-2000	Steady State
Disruptions - ms	0.02-0.3	0.3-20	3-20
Neutron Exposure			
Wall load - MW/M^2	~ 1	1-7	3-6
Fluence - $MW\text{-}y/M^2$	$<< 10^{-5}$	0.02-5	3-10
Surface Materials	TiC coated C C, Be, Mo	Be, BeO, SiC C, Ta, W	Be, W, Ta
Structural Materials	Steels Mo or Ni Alloys	Steels Ni or Cu Alloys	Refractory or Cu Alloys
Coolants	None	H_2O	H_2O, Liquid Metals

As the near term facilities are designed in more detail, the simultaneous neutron damage and high T_2 flux will cause a reassessment of material and design options. The use of high strength copper alloys will probably be one of the first changes for PIC's in near term devices. Also the use of T_2 barrier materials and coatings with low sputtering coefficients will have to be performance tested during high neutron damage levels (up to 50 or 100 dpa). No such simultaneous tests are now scheduled to our knowledge but they certainly will have to be conducted before committing to multibillion dollar facilities.

5. Tritium Breeding for Near Term Facilities

When fusion power levels exceed a few hundred MW and more than a few full power years of operation, the cost of tritium becomes a significant part of the operating costs. It currently costs approximately 10 million dollars to buy a kg of tritium. However, even aside from the costs, the annual availability of T_2 will only be a few kg from CANDU reactors and probably no more than 5-10 kg/y from T_2 producing countries (United States, France, or USSR). Such T_2 availability will not support more than ~ 200 MW_{th}-y of fusion power per year whereas the Canadian T_2 will support no more than 50 MW_{th}-y per year.

Test reactors such as NET and INTOR will try to provide some of the T_2 by breeding it in the least demanding manner possible. Usually that means low temperatures to avoid materials degradation and to reduce the T_2 loss to the coolants. Solid breeders such as Li_2O or $LiAlO_3$ have been considered and tests of these materials in fission reactors around the world show that both the T_2 production rates and the temperatures in the solid breeders can be controlled so they present only a minimal risk to the operation of the test reactor.

The use of liquid breeders such as liquid Li or $Pb_{83}Li_{17}$ have been avoided in near term U.S. facilities because of the elevated temperatures required for circulating the liquid and removing the T_2 (i.e., at least 200-300°C). The liquid metals were also avoided in near term U.S. facilities because of MHD pumping losses. However, NET designs currently include the possibility of using $Pb_{83}Li_{17}$ self-cooled blankets.

A new concept for "cold" breeding of T_2 in near term facilities has been proposed by Steiner et al. [23]. This concept would use an aqueous solution of heavy water (D_2O) and a lithium salt contained in a Zircalloy blanket. The neutronics of such a system is favorable to breeding a significant fraction of the T_2 required when needed. At all other times, light water coolant can be used during the testing and startup phases.

More emphasis will undoubtedly be placed in the future on "cold", partial T_2 breeding blankets which may not extrapolate to a power plant. However, the experience gained with such a technology, coupled with reactor relevant blanket test modules should provide the base for commercial designs.

6. Unsolved and Critical Problems for Near Term ICF Reactors

There has been essentially no effort in the past decade to solve unique materials problems associated with ICF facilities. Aside from surface evaporation due to high heat fluxes [24], two outstanding problems associated with ICF neutrons are evident: pulsed radiation effects and damage to sensitive optical coatings.

The wide disparity between the displacement rates and pulse length associated with fission, magnetic fusion, heavy ion irradiation and inertial fusion facilities is demonstrated in Fig. 5. Whereas fission and (hopefully) magnetic fusion reactors will operate for days or even months at relatively

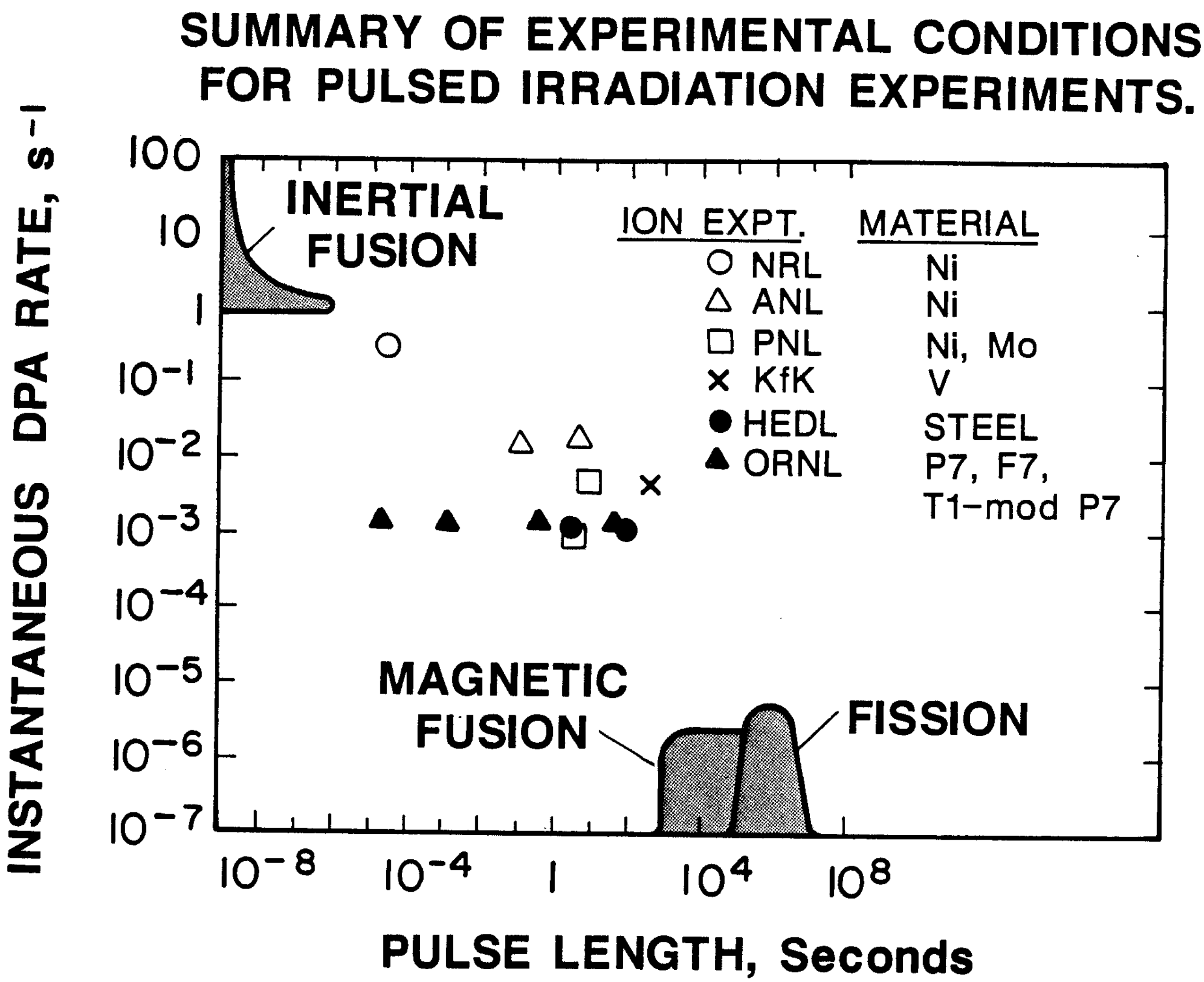


Fig. 5. Summary of pulsed radiation damage conditions present in MCF, ICF and accelerator facilities.

"low" damage rates (10^{-7} to 10^{-6} dpa/s), ICF systems will operate at dpa rates which are a factor of 10^6 to 10^7 higher. It has been shown both theoretically [25-27] and experimentally [28-31] that pulsing, and the time between pulses, can significantly affect the resulting microstructure. Defects such as voids, loops and precipitates can be enhanced or suppressed depending on the operating temperatures [25]. Unfortunately, except for early experiments by NRL scientists [28] which showed a dramatic difference between steady state and pulsed microstructures, no work in the appropriate damage environments has been conducted.

Finally, the current trend toward shorter and shorter laser wavelengths for ICF targets has necessitated the use of dielectric coatings on the reflective optics facing the neutron bursts from the chamber. It is known that such coatings are very susceptible to radiation damage and Fig. 6 illustrates the current laser damage threshold for various glass coatings. The laser damage thresholds vary from 1 to 6 J/cm^2 and values on the order of 5-10 J/cm^2 are required over long periods of time. Even though it is expected that the unirradiated damage thresholds are reduced in the presence of x-rays, neutrons or charged particles, the magnitude of the reduction is not known. Fortunately, the soft x-rays and target debris can be stopped in a few torr-meters of inert gas, but one is still left with considerable damage from the uncollided neutron flux. A convenient conversion factor is 1 rad $\cong 3 \times 10^8$ n/cm^2 (14 MeV). Such a conversion dictates that the final mirrors must be placed 10's of meters away from the target if they are to receive less than 10^{12} rads and last for reasonable periods of time. For example, the final mirrors in SIRIUS-M are placed 20 meters from the target and would accumulate 8×10^{10} rads in 1 FPY year at a 134 MW fusion power level. Placing the mir-

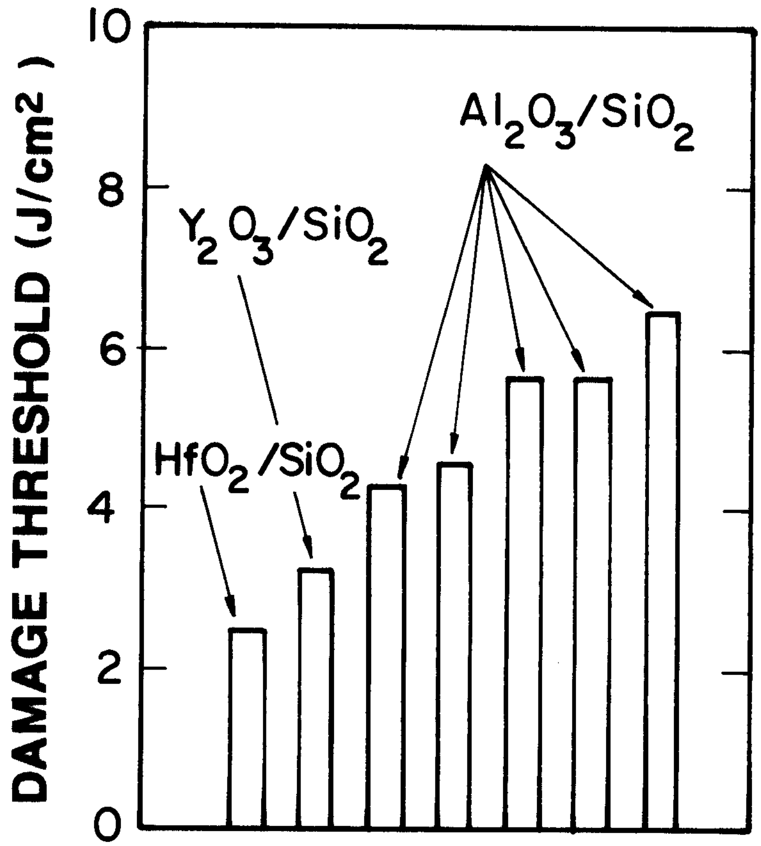


Fig. 6. Range of unirradiated laser damage thresholds for 248 nm reflectors [32].

rors farther away or replacing them frequently will be a heavy economic penalty and therefore a concerted effort must be mounted to find a radiation damage coating material.

7. Conclusions

It is shown that near term magnetic (and to some degree ICF) fusion reactors will not have particularly high damage levels to structural components. Specific components which are more sensitive to radiation effects such as superconducting filaments, electrical insulators, or reflective coatings are likely to be the first to fail in the fusion environment. It was shown that there is a large cost savings if more damage resistant magnet materials can be found. Such programs need to be completed even before solving the structure material problems.

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