Progress in Materials Research for Fusion Reactors

G.L. Kulcinski

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I. Introduction

One of the important conclusions that has emerged from the many fusion reactor studies performed since 1971 is that the long term viability of this energy source is intrinsically coupled to the solution of the materials problem.(1-11) Early studies were overly optimistic with respect to coolant operating temperatures and useful lifetimes that could be attained from rather exotic structural materials . It wasn't until reactor designs appeared with more conventional alloys (e.g., stainless steel) that we began to appreciate the immense problem that lay before us.

Since those "early" days of reactor design a great deal of progress has been made in some **areas**. The object of this paper is to put into perspective some of those advances that have taken place since the 1974 Fusion Reactor Conference in Culham.(1) The format for this discussion is exhibited in Table 1 where we divide the topic into three areas: structural metals, non-metals, and radiation environment. We will then address these topics by specific examples with respect to the degree of progress (or lack of it) and also to list the new ideas which have been proposed since 1974 for the solution of such problems. The reader will certainly recognize a detailed discussion of all the research effort since 1974 is not possible in this short article and therefore, only a few of the more important examples will be discussed here. Finally, a few observations on the use of scarce materials in high powered reactor designs will be made.

Table 1

Status of Fusion Materials - 1977 vs. 1974

	Structural Metals	Non Metals	Radiation Environment
Considerable Progress	. He/dpa in steel . Blistering . n-sputtering . Lifetime Determination	. Solid Breeders (ideas) . Carbon in DT Plasma	. Low Volume Neutron Source . Irradiation Environment (MFR) . Damage Chronology (ICFR) . Radioactivity Calculations
Not Much Progress	. Fatigue, Creep Limits . Void Nucleation . Permanent 1st Wall	. Solid Breeders, Neutron Multipliers (exp) . Insulators	. Pulsed Neutron Source . Pellet Spectra
New Ideas	. Ti Alloys . Lower Operating Temperature . ISSEC's	. Li_2O Coolant . Carbon Curtains	. Tokamak MTR (TETR) . 'Liquid' Fall Protection . Gas Protection

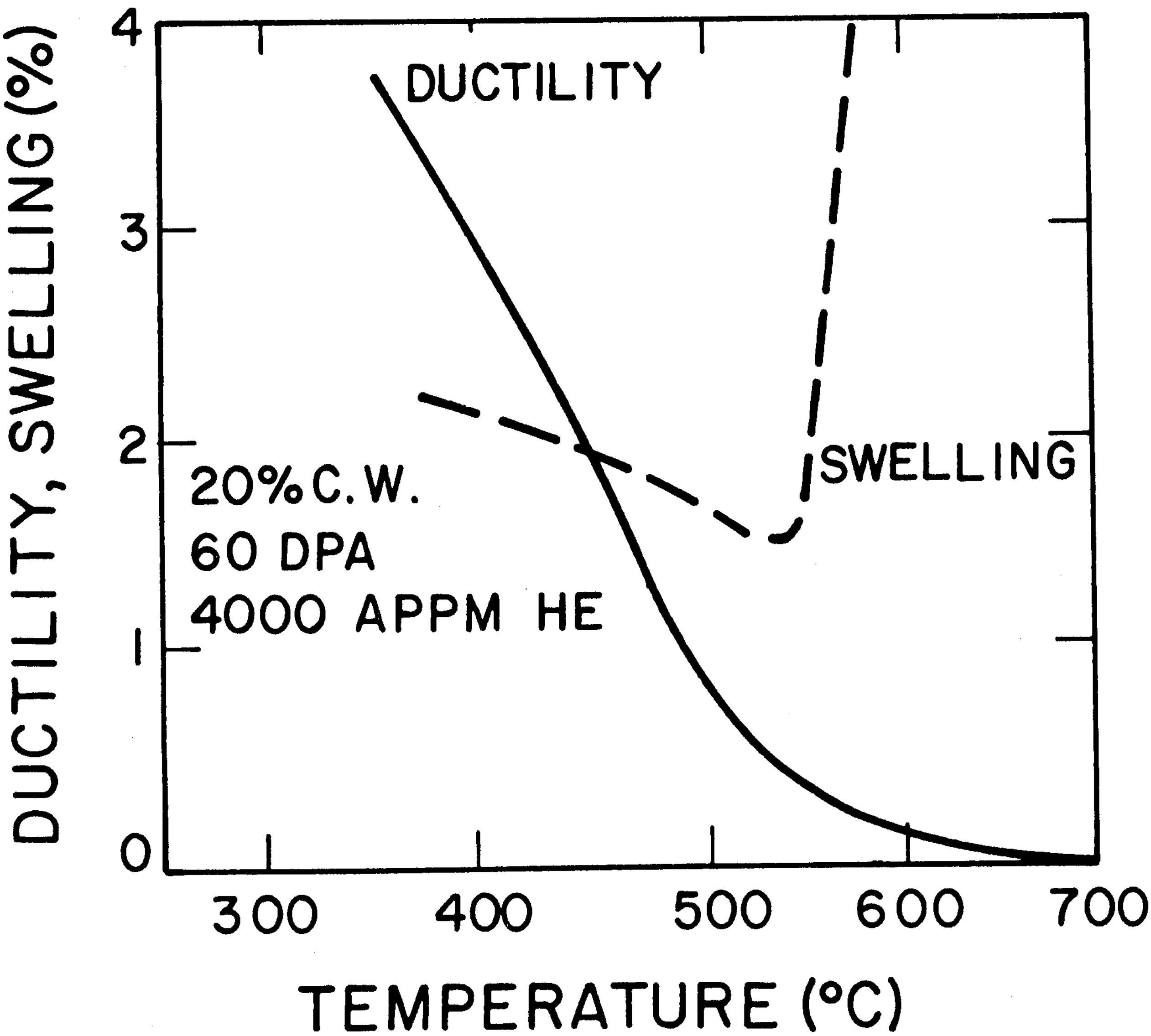
II. Structural Metals

A. Significant Progress

One of the most impressive advances that has been made in this area since 1974 is the simulation of the proper helium to displacement damage ratios in 316 SS. Scientists have used the Ni-59 (n, alpha) reactor to produce as much as 6000 atomic parts per million in 316 stainless steel irradiated over a wide range of temperatures and up to displacement levels corresponding to 8-9 MW-yr/m^2.[(12)] Figure 1 summarizes two of the most important effects noticed in this material. First, the ductility drops drastically with temperature and above 500°C, it appears that useful lifetimes will be severely limited (especially when safety factors are included) as was previously predicted.[(13)] Secondly, the void swelling is greatly enhanced above 500°C by the excess helium and this will also tend to limit the useful lifetime because of swelling gradients. While we now have a much more solid data base from which to predict useful lifetimes for stainless steels, the situation for the other materials (i.e., V, Mo, Ti) has <u>not</u> improved significantly since 1974 because there is no known way to produce helium commensurate with the displacement damage without seriously altering the chemical composition. Until high energy, high flux neutron sources become available in the middle 1980's, we will have to rely on theoretical models, however primitive they are at present.

Another area in which one can see progress since 1974 is that of blistering phenomena. Many more materials have now been investigated

FIG. 1
EFFECTS OF HIGH DAMAGE LEVELS ON SWELLING AND DUCTILITY IN 316SS-ORNL



(e.g., steel, Al, V, Nb, Mo, Be, C)[14-16] over a wide temperature range and with various energy-light ions. Several theories have been proposed and the possible consequences of blistering may not be as severe as originally thought.[21-24] The question of how many exfoliated layers could be lost is still unanswered but the indications are that the thinning rate may not be a constant under a wide spectrum of particle bombarding energies. Furthermore, methods to prohibit blistering altogether have been proposed (e.g., carbon curtains[25] and divertors [1,3,13]). While one cannot say that the problems with blistering are completely solved, the outlook does look brighter now than in 1974.

An area where there was considerable concern in 1974 was that of high energy neutron sputtering. Preliminary measurements indicated that the sputtering coefficients for 14 MeV neutrons were ~0.2 atom/neutron.[21] At that rate, the erosion of the first wall may approach hundreds of microns per year. However, more detailed experiments revealed that such numbers were too high and in fact the sputtering coefficient may be less than 10^{-4} atom/neutron (see Figure 2).[27,28] At those levels, physical thinning of the first wall appears to be no problem.

Finally, our ability to determine useful first wall lifetimes has greatly improved from the preliminary estimates based on uniform ductility loss or gross swelling. We now are including such properties as fatigue, fracture, toughness, creep-rupture, crack propagation, etc., in the analyses of the first wall.[29-31] Figure 3 shows one such analysis performed on an early UWMAK-I design.[19] Unfortunately, the application of more ways

FIGURE 2

EXPERIMENTAL NEUTRON SPUTTERING YIELDS FOR NIOBIUM (AFTER HARLING[27])

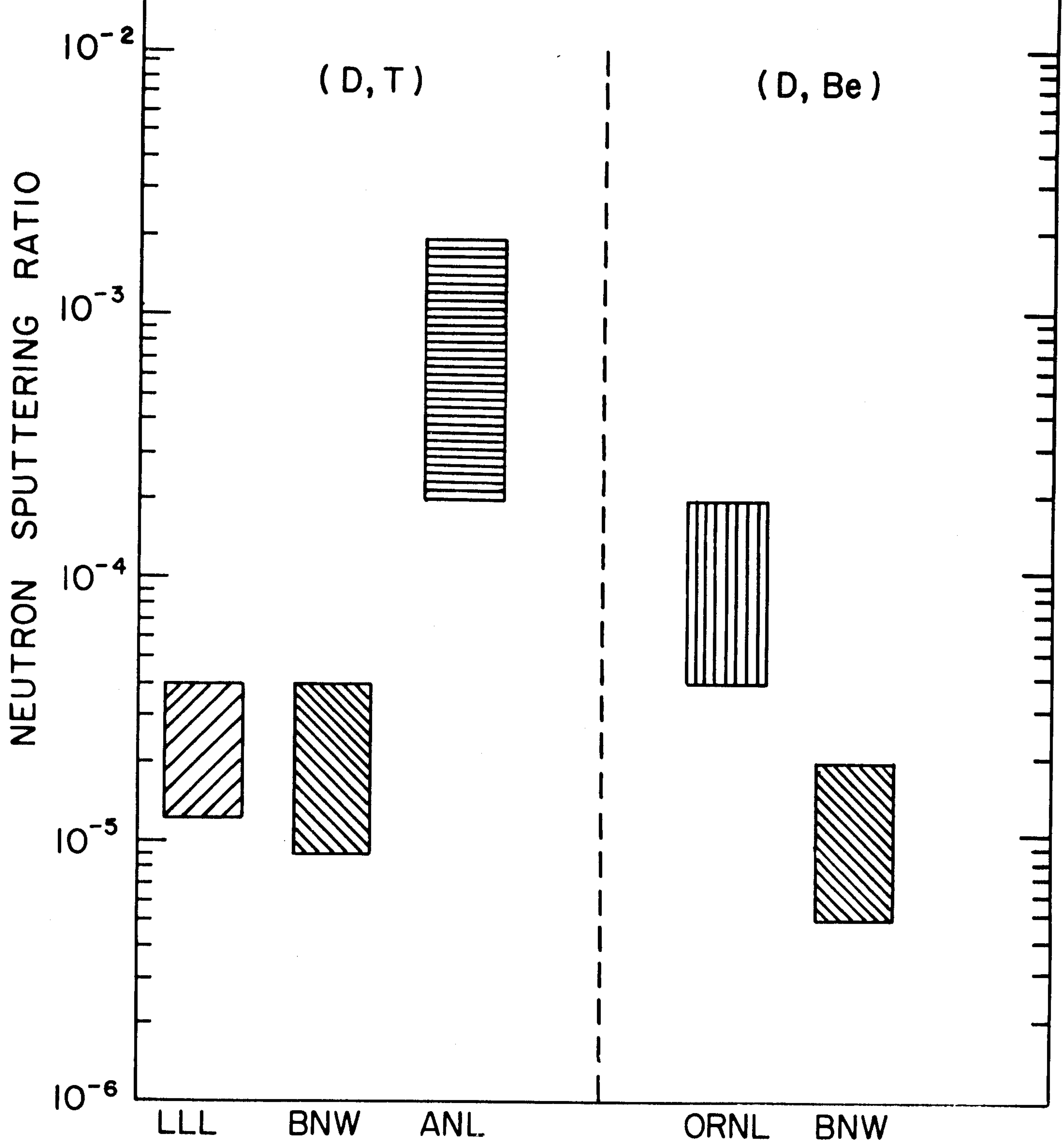
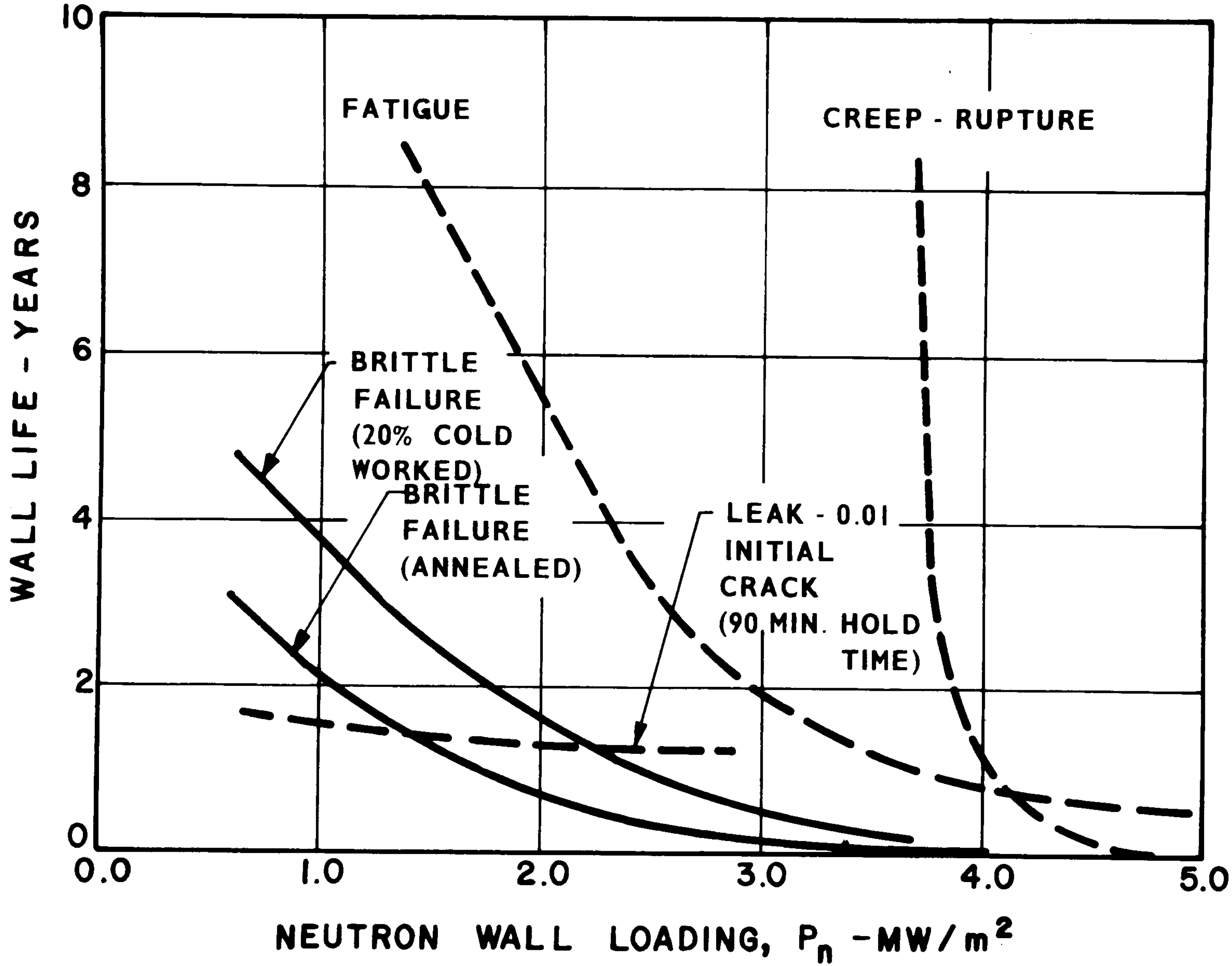


Figure 3

WALL LIFE PREDICTIONS FOR UWMAK-I

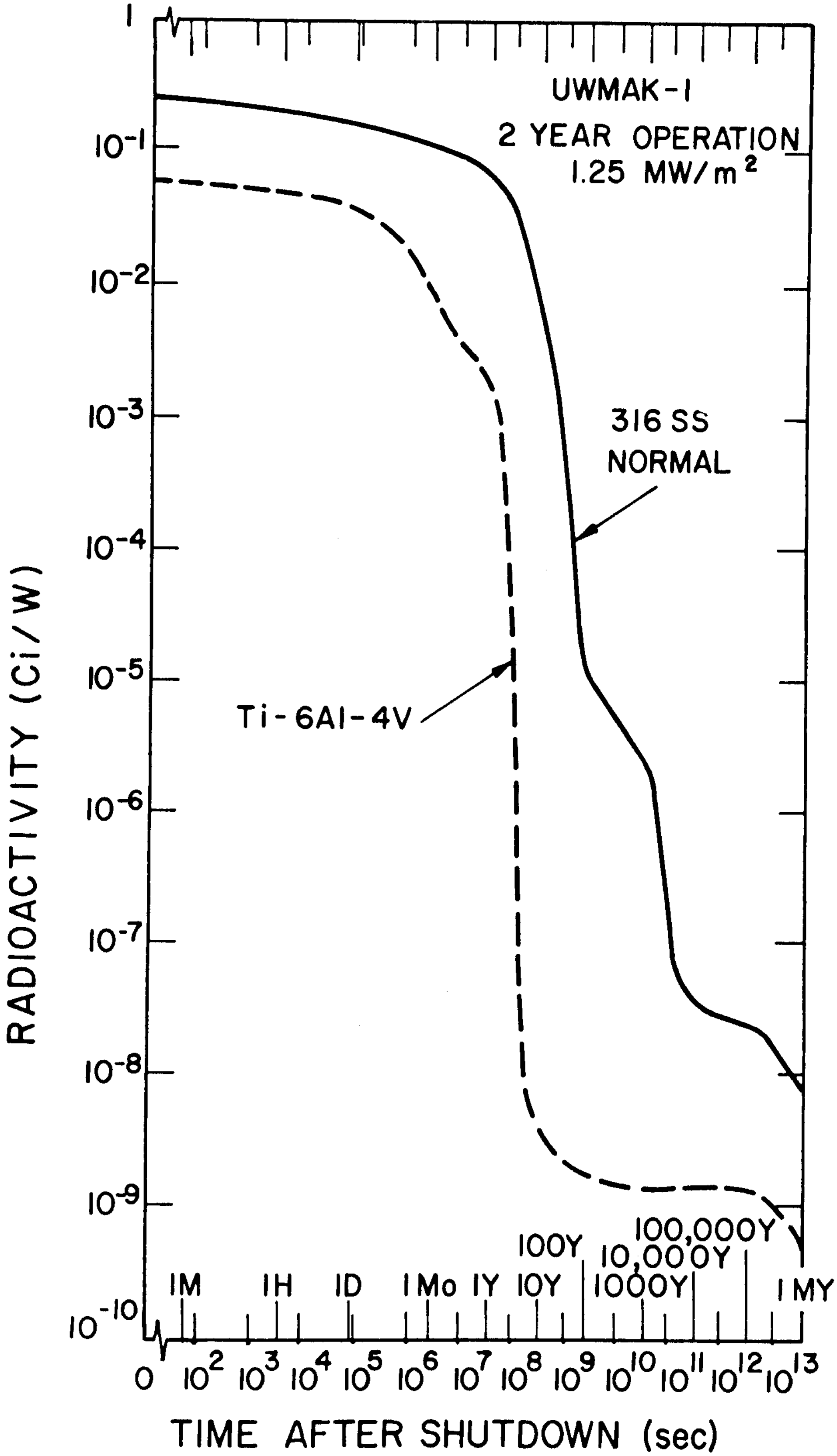


to analyze the wall has not always increased our predicted wall lifetimes. In fact, in some cases it has resulted in lower lifetimes than previously estimated. The main point here is that the broadening and improvement in the assessment mechanisms has revealed further design changes that are necessary and these analyses have strengthened our confidence about achievable wall lifes.

In the area of new ideas, the proposal to use Ti alloys has been made since 1974.(32) This alloy system is characterized by high strength to weight ratios, excellent fatigue resistance, good corrosion resistance, excellent radioactivity properties and the titanium system also indicates a certain degree of resistance to void induced swelling.(33,34) There exists a well-established industry to manufacture 10's of thousands of metric tons of finished products per year and the resource picture looks bright.(32)

On the other side, Ti alloys are more expensive than the steels, a great more of high fluence, high temperature neutron damage data is needed, and hydriding is a worrisome possibility below ~300°C. An example of the low long lived radioactivity of a Ti-6 Al-4 V alloy is shown in Figure 4 compared to a normal 316 SS structure in a Tokamak reactor.(35) It is conceivable that one could actually recycle some Ti alloys within 30 years of plant shutdown(36) thus avoiding the long term (thousands of years) storage of the isotopes in a 316 steel structure. This is a strong psychological advantage for the Ti alloys if nothing else.

There has also been a consistent trend toward lower and lower operating temperature systems in fusion reactor blankets since 1974.(6,7) The drop



RADIOACTIVITY IN 316SS AND Ti 6Al 4V ALLOY DT FUSION REACTOR BLANKETS

Fig. 4

from the 800-1000°C heat of the very early reactor designs to the 500-650°C values of the next generation (UWMAK I[13] & II[37] and PPPL[38]) was the first step. However, it was soon discovered that even this was not satisfactory for Fe-Cr-Ni alloys and more recently temperatures of ~400°C[6,7] have been proposed. This lowering of the coolant outlet temperature also greatly relaxes the balance of plant design as well as allowing a possibly longer wall life to be enjoyed. Other indirect benefits are the lowering of the tritium diffusion rates and the use of simpler materials in the rest of the power cycle. The price that had to be paid for these benefits is lower efficiency and more thermal pollution potential. However, it appears that even at these lower temperatures, the fusion reactor will be as good as the LWR's in this respect.[7,8]

Another idea which has capitalized on the flexibility of the fusion process and shows great promise for increasing the first wall lifetimes is the ISSEC (Internal Spectral Shifter and Energy Convertor) concept.[39-42] The basic idea here is to place a passive shield between the plasma and the first vacuum wall to slow down the 14 MeV neutrons, extract their kinetic energy and radiate the heat to the wall. Such a shield in itself would operate at very high temperatures and probably would be made out of carbon to withstand the intense heat. Calculations have shown that such a C ISSEC may operate at ~2000°C where most of the displacement damage would simply anneal out and the helium could diffuse from the shield.[42] An example of the neutron spectra with and without a 25 cm carbon ISSEC is given in Figure 5. Note that the high energy neutron component is reduced by more

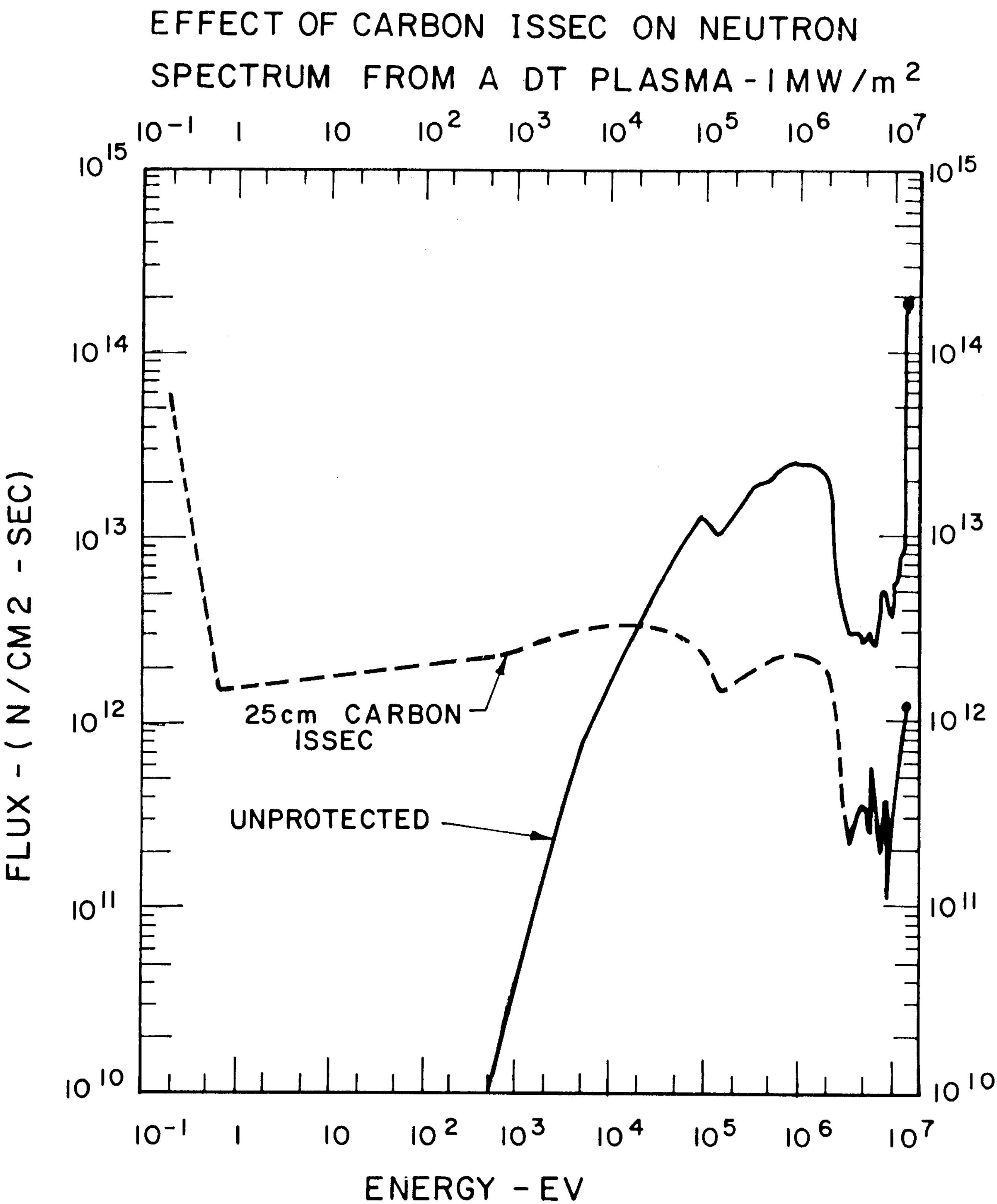

EFFECT OF CARBON ISSEC ON NEUTRON
SPECTRUM FROM A DT PLASMA - I MW/m²
FLUX - (N/CM2 - SEC)
ENERGY - EV
25cm CARBON
ISSEC
UNPROTECTED

Figure 5

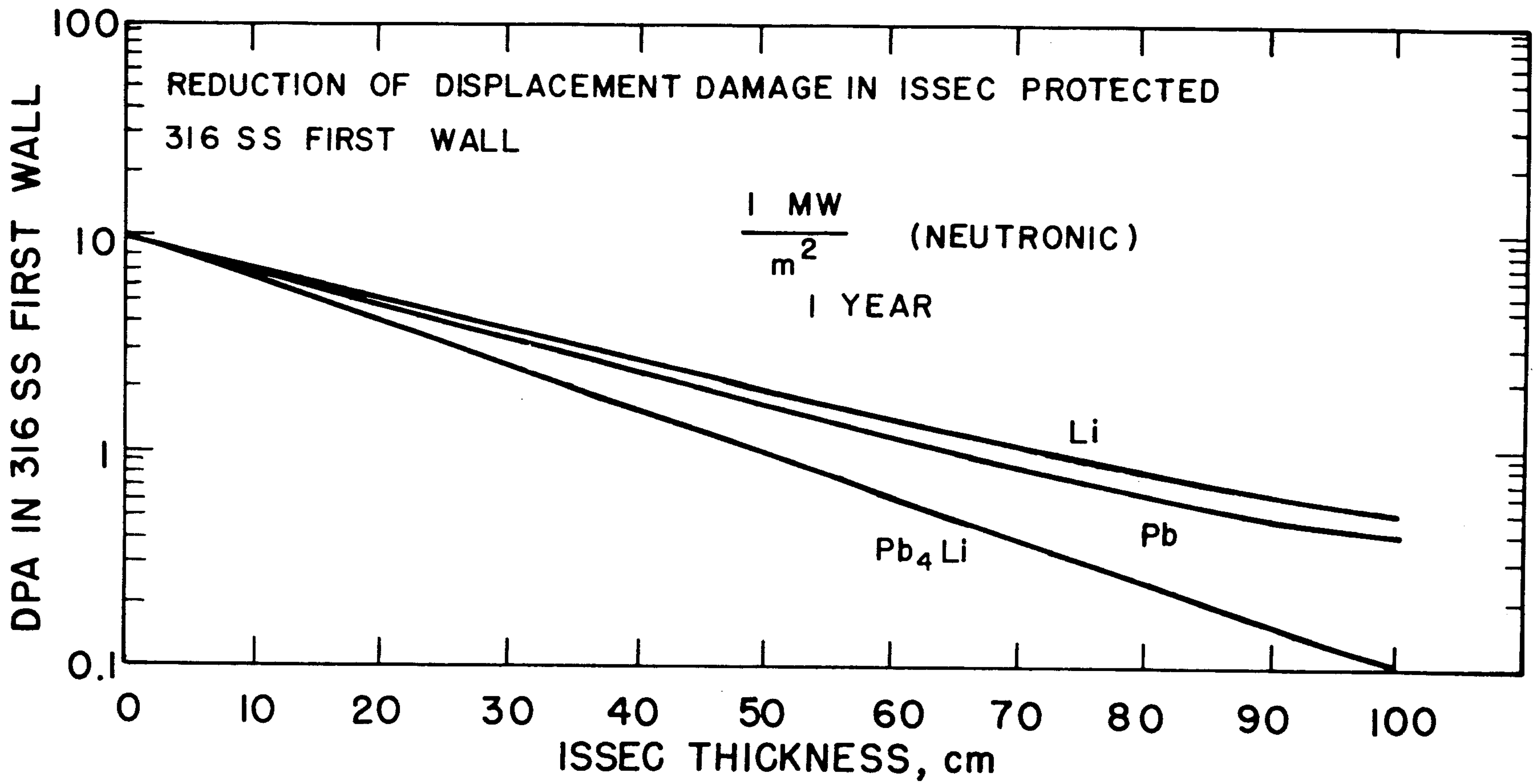
than 2 orders of magnitude and the thermal flux is greatly increased. This change in neutron spectra results in a large decrease in the displacement damage and helium production rate.

This concept has been recently applied to the Lawrence Livermore Laboratory laser fusion reactor design[43] in the form of a liquid Li "waterfall".[44,45] (See description in Section IV). This liquid ISSEC concept also can reduce the damage rate in the structure (see Figure 6) and the moving Li carries the heat away instead of radiating it to the first wall. The real importance of this concept lies in the order of magnitude or more reduction in damage level incurred by the vacuum containing structure and in fact it may be our only hope to obtain a permanent (~100 MW-yr/m^2) first wall structure.

B. Areas Where Significantly More Effort is Required

A key feature of all the recent fusion reactor designs (except for the mirror) is the pulsed nature of operation. This has always been appreciated for the inertial confinement schemes and the Theta Pinch, but even the Tokamak reactors now appear to present problems in this area. The desire to minimize the temperature and pressure transients in these systems has pushed engineers into designing short down times between "burns". Furthermore, power supply limitations have tended to shorten the "burn" time itself such that it is now not uncommon to discuss burns of a few minutes followed by 10 seconds or so of down time for impurity removal, resetting of magnet power supplies, refueling, etc.[7-8] Such cycles mean that at a 70% plant factor, approximately 100,000 transients will take place in the blanket per year and over 3 million cycles will be incurred

Figure 6


REDUCTION OF DISPLACEMENT DAMAGE IN ISSEC PROTECTED
316 SS FIRST WALL
1 MW/m2 (NEUTRONIC)
1 YEAR
Li
Pb
Pb4 Li
DPA IN 316 SS FIRST WALL
100
10
1
0.1
0
10
20
30
40
50
60
70
80
90
100
ISSEC THICKNESS, cm

over the plant lifetime. Even though the problem has been recognized, very little in the way of stress or strain cycles can be calculated until detailed designs are presented and complex analyses of joints, weld zones, and diagnostic ports are performed.

On the materials side, very little has been done to find materials capable of withstanding such severe cyclic loads. Experimental measurements of fatigue life have only been performed on a small number of alloy systems and there have been essentially no tests of these alloys in a reactor environment. Theories to predict the useful fatigue life in the presence of hydrogen isotopes, liquid metals, neutron irradiation and rapidly changing loads have not been developed yet and do not appear to be a major part of the effort in the fusion materials' community. This will have to change in the next few years.

The situation with respect to thermal creep is much better now than in the early reactor designs where the structural metals were operated at $\sim 0.5\ T_m$ (where T_m is the absolute melting point). Now the reactor designs have tended toward lower blanket temperature ($\sim 0.3\ T_m$) and thus have removed thermal creep as a major factor.[6,7] However, the irradiation induced creep is still present and in fact, takes on much more importance at the lower operating temperatures. Irradiation induced creep can relax the stresses induced by coolant pressures, magnetic fields, or the maintenance of a vacuum during the "burn" but these stresses will reappear when the neutrons are "turned off" again and the driving force for the induced stress is changed.

Irradiation creep studies are very expensive to run in a reactor environment and to date only a few such successful studies have been performed on fission reactor materials.[46] No such studies have yet been performed in the unique fusion environment of any reactor system. Furthermore, theoretical models describing this process have not been developed to the satisfaction of most of the materials community. Preliminary examination reveals that the time between the "burn" in a Tokamak reactor (when the relaxed stresses can reappear) will be critical for the long term survival of first wall materials and could well represent the most critical period for the first wall.

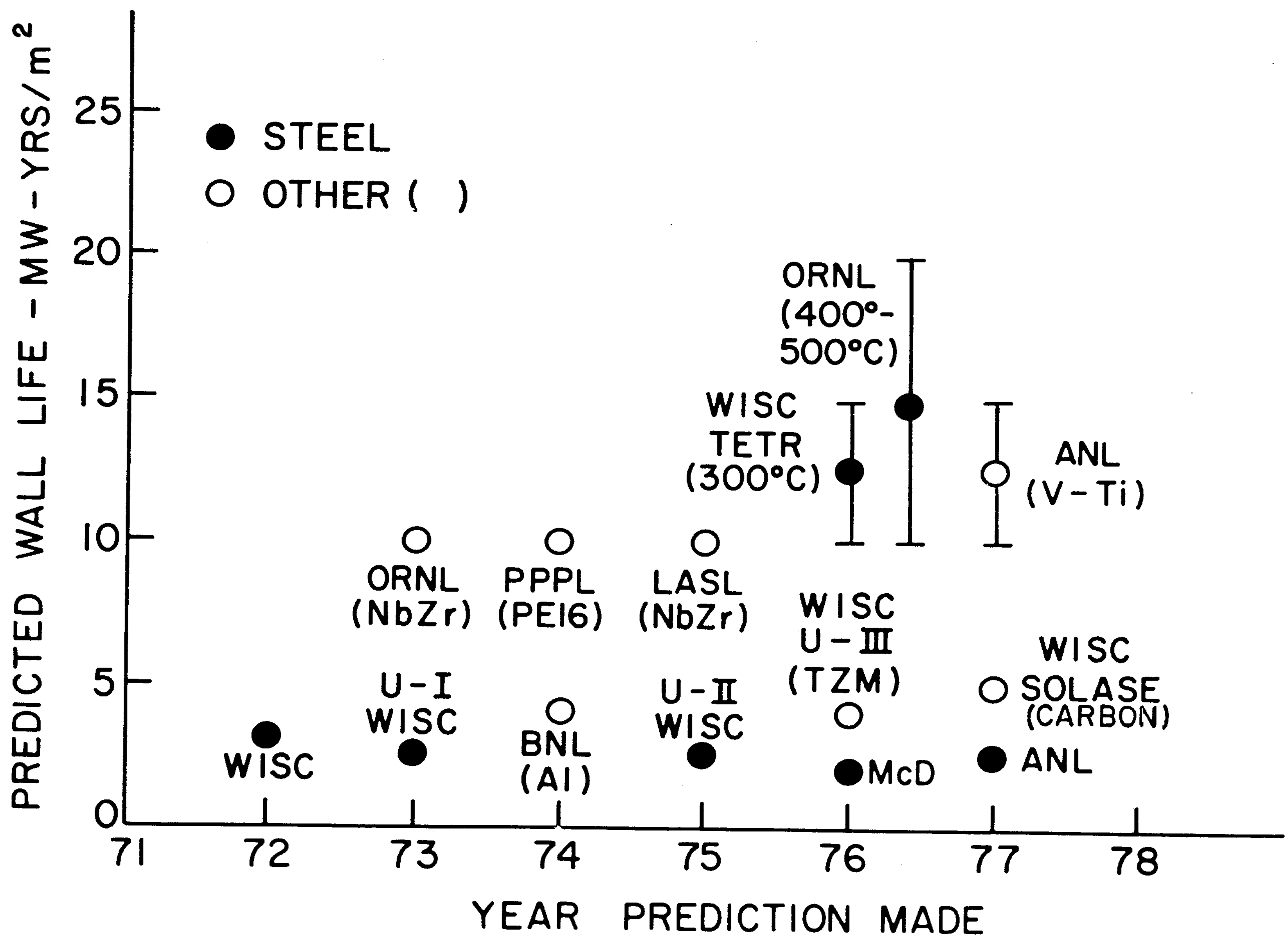
It has been shown by several theoretical[47-48] and experimental[49-50] studies that void nucleation is extremely sensitive to impurity atoms, especially inert gases.[51,52] These latter impurities are a common transmutation product in a DT fusion neutron environment and since most blanket designs are in the temperature range where void swelling will be a problem, an understanding of this problem is essential. Void nucleation theory is a very complex field as witnessed by the fact that there are only 3 or 4 individuals or groups in the world working in the area. Nevertheless, since the problem of void swelling is expected to be critical to the continued operation of structural components, much more effort needs to be invested in this area to understand and beneficially control this phenomena.

The probability of having the first wall of any CTR design last the entire (~20 full operating years) lifetime of the reactor seems to be no more at hand now than in 1974. This is due to two trends. First, the trend has been toward higher wall loadings (4-6 MW/m^2 in high beta tokamaks without divertors)[7,8] and this means that one would require ultimate lifetimes of ~100 MW-yr/m^2 (6 times the maximum damage level projected for the cladding in the highest burnup fuel element in an LMFBR). Secondly, we have become much more quantitative in our ability to analyze potential failure modes for fusion reactor first walls and this has tended to restrict the potential for dramatic improvements based on a single material property (e.g., fatigue, yield strength, ductility, etc.). For example refer to Figure 3.

Figure 7 gives a brief summary of the predictions of first wall life for various materials. Note that the wall life of steels above 500°C is predicted to be less than 4 MW-yrs/m^2 and below 500°C it may reach 10 MW-yrs/m^2. The values for the Nb, TZM, and V alloys are only speculation at this point as there is very little data on which to base such predictions. The important point of Figure 7 is that we are still an order of magnitude away from the first wall lifetime that we would like to have for a complete reactor lifetime. Accepting the fact that we will have to periodically change the blanket structure has a large impact on capital costs (through increased remote handling equipment and hot cells), operating costs (replacement parts, increased down time) and environmental concerns (through increased radioactive waste quantities and demand on scarce resources).

FIGURE 7

PREDICTED FIRST WALL LIFE
VARIOUS REACTOR DESIGNS



III. Non Metals

A. Significant Progress

Several new ideas with regard to solid tritium breeding compounds have been proposed in the past four years. The main improvement has been in the area of higher lithium atom densities (such as in Li_2O)[53,54] and more extensive consideration of Pb-Li alloys.[8,55] Both of these materials offer the potential for breeding ratios greater than 1.0 without the use of Be. Original reactor designs which contained solid breeder such as Li AlO_2[37] required neutron multipliers like Be but it was subsequently shown that the resource problems associated with that element were too formidable to overcome.

Much more interest is now being shown in carbon liners[25] or coatings[56,57] to reduce the contamination effects so prevalent in Tokamak plasmas. Original fears about poor vacuum properties were shown to be groundless[58] and the high diffusivity of helium from carbon above 800°C[59] has removed fears of large dimensional changes due to bubble formation. The rate of chemical sputtering by hydrogen isotopes has been shown to be tolerable above 1000°C[60] and current studies of possible C_2H_2 formation above 1500°C are in progress.[61] Neutron radiation damage data above 1000°C reveals a lessening of the acccumulated damage[63] and very high temperature tests (1500-2000°C) are currently being planned.[64] If these latter tests prove that the dimensional stability is adequate, then the use of carbon liners, ISSEC's, coatings and limiters can be seriously considered for power reactors.

B. Areas Where More Effort is Required

On the negative side for the non metallic materials, very little progress has been made in determining the irradiation effects on solid breeders and neutron multipliers at typical operating temperatures. Since the helium gas production rates in these materials can be very high (~10's of thousands of at. ppm per MW-yr/m^2), swelling and mechanical integrity are real questions.[9] Such tests can be easily performed in existing fission reactor facilities and need not wait for high energy neutron facilities.

The criteria for useful lifetimes and operating conditions for electrical insulators has not been made much clearer in the past four years. One area which is much more prominent now is that of electrical insulators for magnets. There has been a tendency to bring the magnets closer to the plasma to reduce stored energy and power requirements[7,8,37] Such designs now require the electrical insulators to withstand 10^8 rads/year (or more) when leakage down divertor slots and other reactor penetrations is taken into account. These values are close to the recommended design limits for some of the more damage resistance fiberglass insulators[65] and unless periodic replacement of the magnets is contemplated, more radiation damage resistant insulators must be developed.

C. New Ideas

One important new idea to surface since 1974 is the use of solid Li_2O as a coolant.[54] There is a subtle, but very important concept involved here which takes advantage of the unique features of the DT reaction. Flowing beds of ~100 micron diameter Li_2O particles can transport (not transfer) the heat and tritium outside of the reactor where it can be recovered in a timely fashion. The neutrons pass through the first structural wall and slow down in the Li_2O as well as reacting with the Li-6 to form tritium. The exothermic reactions then deposit the heat in the particles and because of the poor heat transfer coefficients from particle to particle,the heat gets transported out of the reactor to a heat exchanger. These poor heat transfer characteristics allow one to effectively decouple the temperature of the coolant from the structure so that the vacuum containing material can be at much lower temperatures.

The use of carbon "curtains" to protect the plasma from high Z impurities sputtered from the vacuum chamber walls is also a new concept proposed since 1974.[25] Flexible carbon cloth is a product well-known in the space industry and it has also been calculated that the hanging of this cloth between the plasma and first wall can protect the wall from sputtering by ions and neutrals escaping the plasma and from localized plasma dumps in the case of abnormal operation. It has also been found that the heat capacity of these curtains is high enough to reduce the first wall temperature fluctuations between the burns.[6] Operation at 1000-1200°C appears to be reasonable from a strength and chemical erosion basis as well.[42]

IV. Radiation Environment

A. Significant Progress

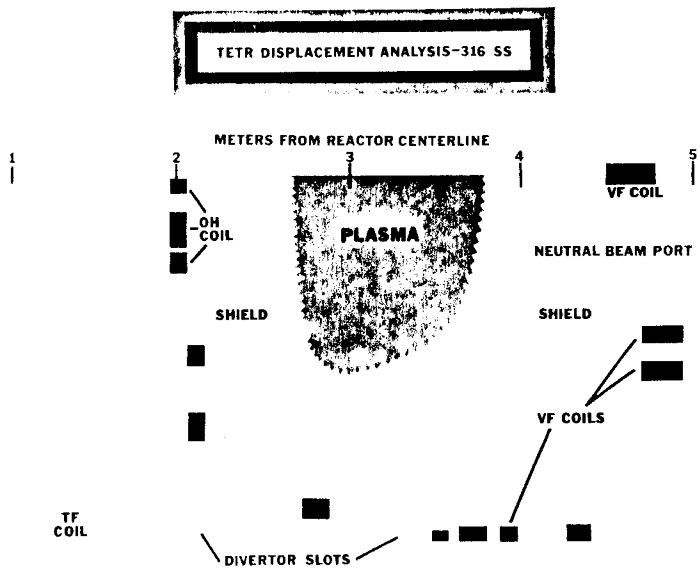
One of the important measures of progress in this area since 1974 is the authorization and construction of several high flux, high energy neutron sources to test materials. The solid target RTNS-II[66] has been designed and is currently under construction at LLL in the U.S. (See Table 2 for status). It is expected to operate in March 1978. The Intense Neutron Source at LASL[67] is ~25-30% completed with a target date for operation of October 1979. Finally, the construction of a high energy D-Li source, a concept orginally conceived at BNL,[68] has been approved at the Hanford Engineering Development Lab and it is expected to be in operation in late 1983. All of these sources have low volume test areas (~1 to several hundred cm^3 of specimen test volume excluding temperature and load control equipment) and they should provide limited experimental verification of the theoretical predictions for radiation damage in a DT neutron environment. However, they are not adequate for a full fledged alloy development program.

The description of the irradiation environment to be experienced in Magnetic Fusion Reactors has come a long way since 1974. In particular, neutronics calculations have progressed from the 1 dimensional-homogenized slab geometry consideration to more realistic 3-dimensional discrete blanket analyses.[6,70] An example of such a calculation is shown in figure 8 for the blanket-neutral beam port section of the TETR reactor design. We are now able to show how such response functions as displacement rate, transmutation rate, etc., vary angularly as well as with depth. Such calculations can highlight "hot spots" near reactor penetrations where additional shielding may be required. These analyses will also help in determining which parts of the blanket may reach a maximum allowed damage level first and thereby help to more clearly define useful lifetimes.

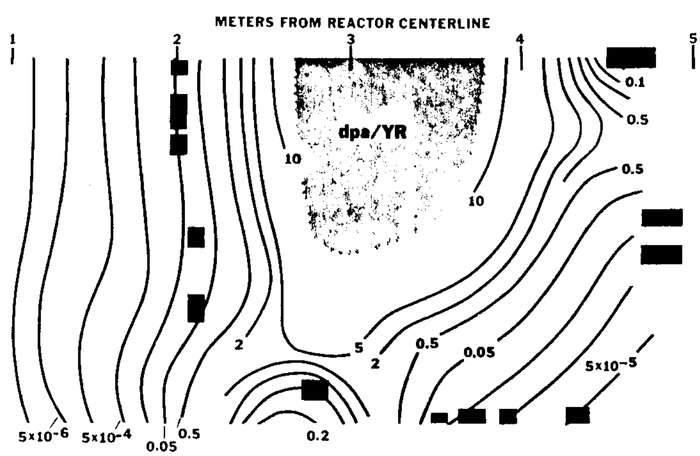
Table 2

Near Term High Energy Fusion Materials Irradiation Test Facilities

Source	Neutron Flux $cm^{-2}s^{-1}$	cm^3 of Test Volume	Status-Nov, 1977	Expected Operation Date
RTNS-II	2×10^{13}	1	Construction 90% Complete	March 1978
INS	10^{14}	3	Prototype Operating Final Design 25% Complete	October 1979
D-Li	10^{14} 10^{15}	300 10	Design Funds Approved FY-78	Late 1983



(a)



(b)

Figure 8 - Displacement Damage Analysis of 316 SS Structure in TETR.
a) Schematic of blanket, shield and divertor with appropriate magnet position and neutral beam ports, b) Lines of constant damage rate (dpa/yr) in the 316 SS.

Another area of significant progress is the determination of the time and spatially dependent damage state in the first walls of inertially confined fusion reactors. It has been shown that the debris from "typical" laser fusion pellets can result in very high thermal, stress, and displacement pulses in the first solid surfaces facing the pellet. Figure 9 shows what the composite temperature and displacement response of a Cu surface would be from a 100 MJ pellet exploded 7 meters away. The detailed pellet spectra is discussed elsewhere[71,72,73] but it is important to note that the temperature at the surface will be quite high before and during the arrival of most of the pellet debris. One example of how such analysis can help in the design of the wall has to do with sputtering. The previously described environment means that sputtering will take place at high temperatures where the sputtering coefficients are much higher than those determined at reactor ambient temperatures[74] and hence the thinning rates will be much larger. Other examples of how important these transient analyses can be, have to do with blistering due helium ion injection, annealing of voids and loops between pulses.

An additional area in which great progress has been made is that associated with the induced radioactivity in the reactors and the impact of such radioactivity on reactor designs. First of all, our ability to quickly, cheaply, and accurately calculate the induced radioactivity has improved with the development of specific computer codes for this area.[75] Next the inclusion of all alloying elements and impurities into the calculations has tended to reveal the true magnitude of background radiation levels.[76,77] Finally, a general appreciation of the difference between safety (short times after shutdown ~ hours to day), maintenance (days to weeks) and long term storage (100's to thousands of years) has been better understood.

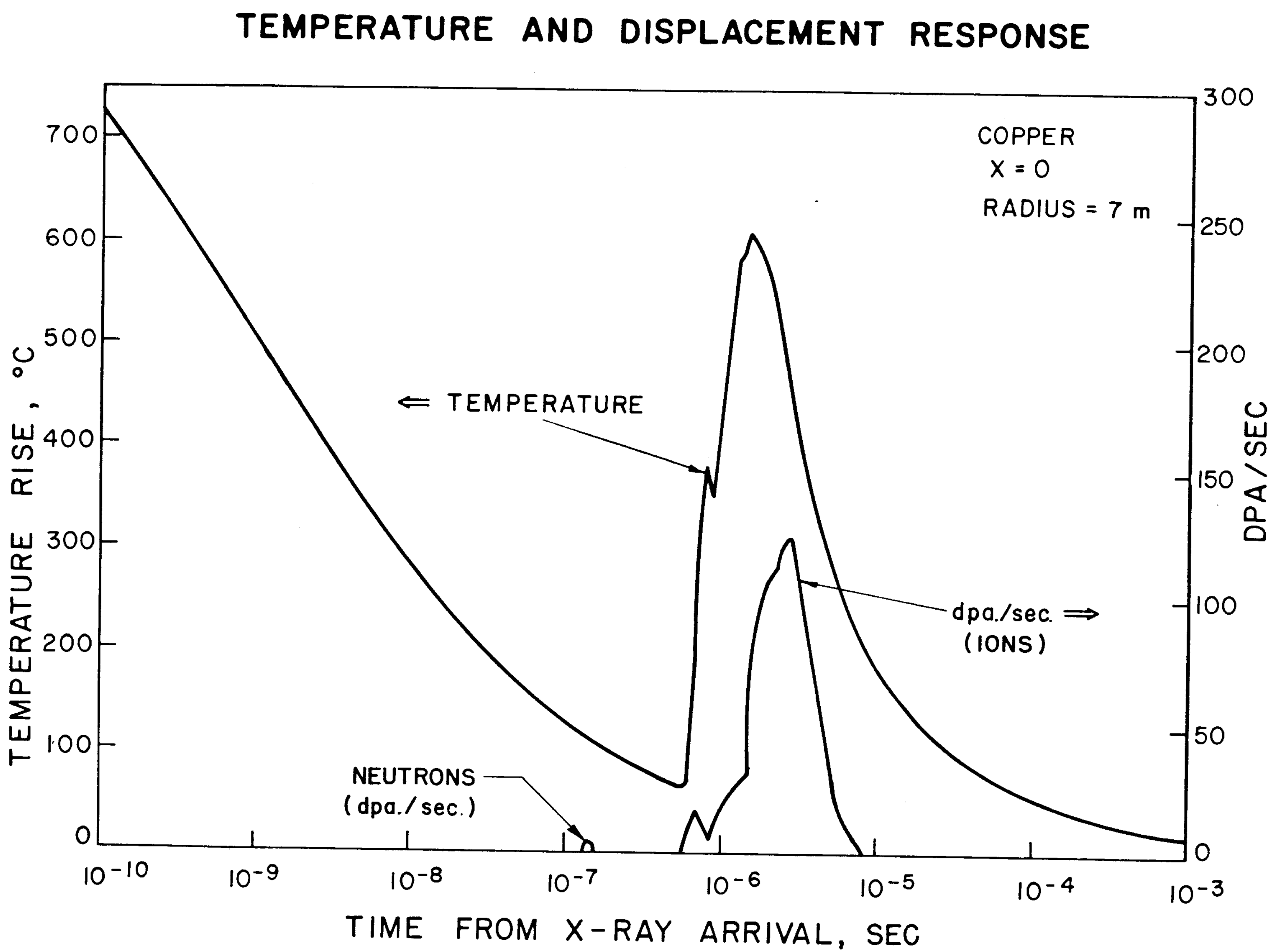

TEMPERATURE AND DISPLACEMENT RESPONSE
COPPER
X = 0
RADIUS = 7 m
⇐ TEMPERATURE
dpa./sec. ⇒
(IONS)
NEUTRONS
(dpa./sec.)
TEMPERATURE RISE, °C
DPA/SEC
TIME FROM X-RAY ARRIVAL, SEC
700
600
500
400
300
200
100
0
300
250
200
150
100
50
0
10^{-10}
10^{-9}
10^{-8}
10^{-7}
10^{-6}
10^{-5}
10^{-4}
10^{-3}

Figure 9

It now appears that all known structural alloys (including C) represent a safety hazard if released outside the reactor during a catastrophic accident.[77] It also appears that hands-on maintenance within a reasonable time of shutdown can be performed on the exterior of the reactor if it is properly designed but that hands-on maintenance of the inner blanket is not feasible at this time.[6] Finally, there are great differences in the long term storage time of the proposed alloys ranging from possible reprocessing in 30 years for some alloys to permanent (>10^4 years) sequestering of others.[77]

B. Areas Where More Effort is Required

Two areas which have not displayed much improvement over the 1974 period have to do with the inertial confinement area. Even though we now know that the displacement rates by neutrons may approach 1-10 dpa/sec for 0.1 microsecond[73] there are no facilities which can come within a factor of 1000 of this damage rate on a repetitive basis. Furthermore, dpa rates of 10-1000 per sec can also occur for 10's of microseconds in the first wall.[73] The highest sustained dpa rate produced thus far is ~0.1 dpa/sec.[78] Obviously if ICFR's are to be seriously considered, experimental facilities to simulate these effects will have to be constructed.

The classification that has surrounded the pellet design has made any specific first wall selection criteria very difficult to formulate. At the present time only spectra from bare D-T pellets is available but it is widely known that these are greatly different from the highly structured, high performance pellets needed for a commercial system. If materials scientists are to develop damage resistant materials, they will have to know what the irradiation environment looks like in much greater detail than presently available in the unclassified literature.

C. New Ideas

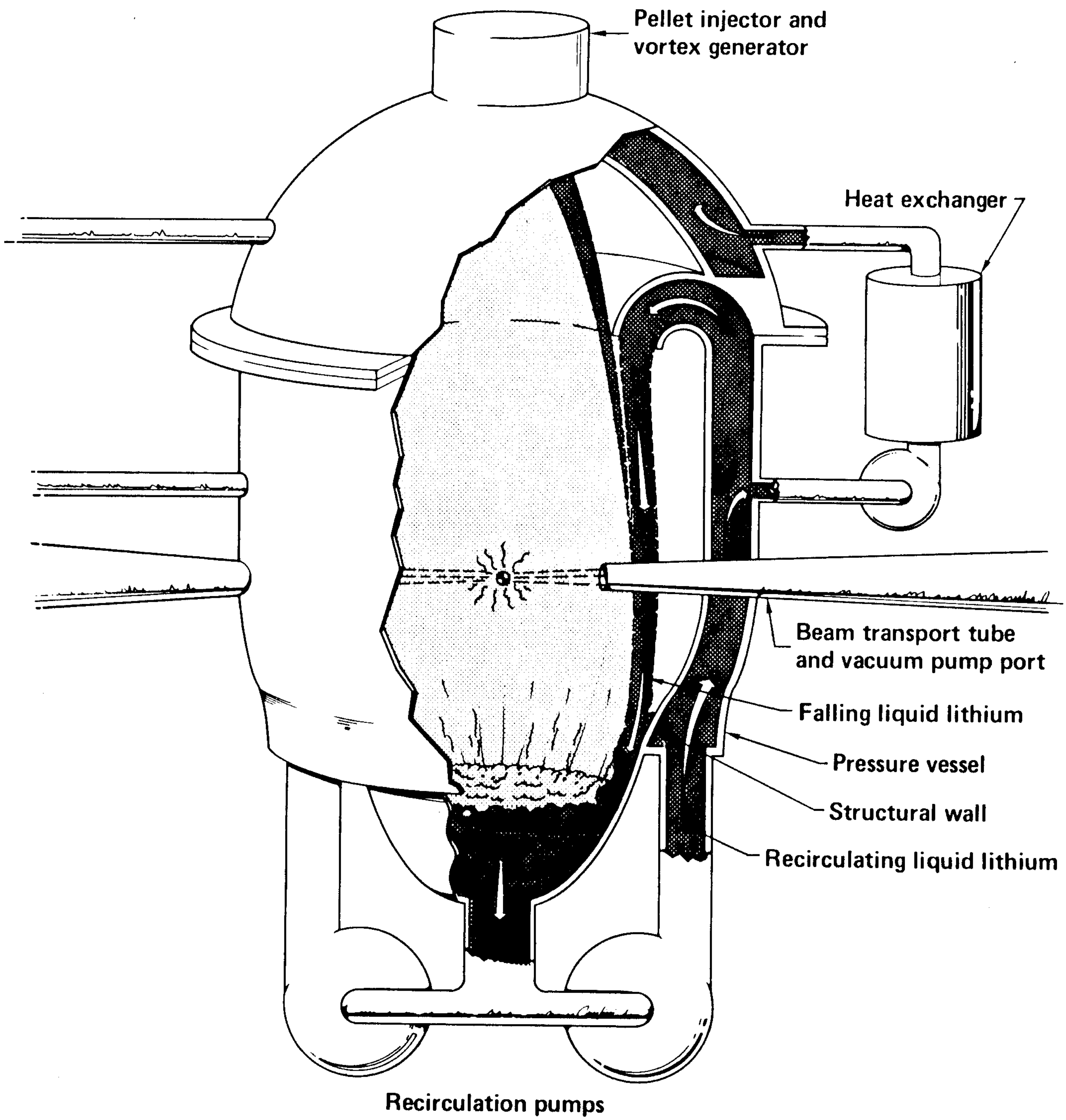
One new idea which has surfaced since 1974 is that of a Tokamak Materials Test reactor.[6] Previously only a Mirror Materials test reactor had been proposed.[79] The main object is to obtain a large ($\sim 10^6$-$10^7 cm^3$) useable test volume in order to screen several alloys at a variety of temperatures, stresses, and coolant environments. Such a large volume facility would necessarily come after the low volume neutron test facilities previously described. The Tokamak Engineering Test Reactor (TETR)[6] would provide 1-2 x 10^6 cm^3 of high flux (>1MW/m^2) test volume. This reactor is based on near term technology and TFTR state of the art plasma physics. It could possibly be in operation by the end of the 1980's and provide the necessary information for the US-DEMO reactor scheduled to operate in the late 1990's.

There have been two new methods proposed to mitigate the first wall surface damage in an ICFR. The first is the "lithium fall" approach proposed by LLL.[43] A schematic of this concept is shown in figure 10. A stream of liquid lithium is dropped in a cylindrical chamber such that when the microexplosion takes place, the liquid absorbs the photons and charged particles. It has even been shown that if the Li fall is thick enough, the moderation of the neutrons can reduce the damage to the first wall components.[44,45] If this concept works, it could go a long way to reducing the complexity of the ICFR design and may even result in a lower cost system.

Another approach to the same problem is to fill the chamber with a low pressure ($\sim$1 torr) high Z gas which will absorb most of the X rays and charged particles before they hit the first walls.[10] This would allow most of the kinetic energy of the pellet debris (except the neutrons) to

Figure 10

LIQUID LITHIUM "WATERFALL" CONCEPT



heat the gas. The exhaust of the high temperature gas and the resulting shock wave to the first wall are problems which should be easier to handle than the surface damage from an unprotected wall.

V. Power Density Considerations for Materials.

The power density of a particular reactor is often quoted as a measure of its ultimate cost. However, several definitions have been used in the past and it is worthwhile to be very specific about them here.

Plasma physicists often define the power density of a reactor as the thermal power of the reactor divided by the volume of the plasma (since the boundary of the plasma is hard to define, the volume of the vacuum chamber is usually used). For the case of laser fusion the volume of the blast chamber is a convenient measure. A power density as determined by either of the above methods can be misleading because while it is indicative of the size of the reactor, one does not have to pay directly for the vacuum volume!

Another method of determining the power density in a DT fusion reactor would be to divide the total power generated by the volume of the blanket required to slow down and extract the energy from the X rays, ions and neutrons. This is called the "neutronic" power density. The drawback with this method of estimating cost is that it does not reflect the cost associated with the confinement or sustainment of the plasma. Items such as magnets, neutral beam injectors, lasers, electron beam generators, etc., also need to be included in the overall cost.

Another way to calculate a meaningful power density is to include the volume of the blanket, shielding, auxiliary equipment such as magnets, lasers, etc.. This would be more properly called an engineering power density. Such a number should be limited to only those items associated with the nuclear island as the balance of plant items should be common for most of the fusion approaches.

Table 3 lists the various power densities for 6 recent tokamak reactor designs, 2 mirror designs and one laser fusion reactor. Note that within the tokamak reactors there has been a steady increase in the plasma power density from 0.6 MW_t/m^3 in UWMAK-I to ~9 MW_t/m^3 in NUWMAK. The Reversed Field Mirror concept has even a higher plasma power density at 17 MW_t/m^3.

One can see from Table 3 that the power density in the blanket can be quite different than the power density in the plasma. For example, the neutronic power density in UWMAK-I is 3 MW_t/m^3 vs. a plasma power density of 0.6 MW_t/m^3. The power density in the NUWMAK reactor increases to 13 MW_t/m^3 and it is as high as 11 MW_t/m^3 in the SOLASE laser reactor design. Other designs such as the Mirror reactors are reduced to 2-3 MW_t/m^3.

Finally, when all of the auxiliary and driver components of the nuclear island are included we see a much different picture. At the present time the variation from the earliest to the more recent reactor designs is only a factor of 3 to 4 and the Mirror reactors show lower engineering power densities, around 0.5 MW/m^3. Even the laser fusion reactor is in the 2 MW_t/m^3 range.

The significance of the above numbers can be appreciated by the following rough calculation. Assume that the average solid density of the

Table 3

FUSION REACTOR POWER DENSITIES

	MW_t/m^3		
	PLASMA	NEUTRONIC	ENGR
UWMAK-I	0.6	3	0.7
UWMAK-II	0.6	2	0.6
UWMAK-III	2	4	1
PPPL	2	4	1
ORNL-DEMO	5	5	2
NUWMAK	9	13	2.5
TANDEM MIRROR	4	2	0.5
FIELD REV. MIRROR	17	3	0.4
SOLASE	4	11	2

material (including 10% coolant) for the reactor is 7 tonnes/m^3 (steel ~8, copper ~9, refractory metals 6-10 tonnes/m^3). If the average fabricated cost of the reactor components is 15 \$/kg (1978 \$) then at a 1MW_t/m^3 engineering power density the nuclear island direct costs are \$315/kWe (at a 33% efficiency). Typically, nuclear island costs are one third of the total direct costs of the reactor plant(8,13,37) and this yields 950 \$/kWe in direct capital costs. The indirect costs of any power plant range from 100-150% of the direct capital costs(80) which means if we use an average of 125%, the total capital costs will be in the 2150 \$/kWe range. This is about twice the current costs of electrical energy and it is clear that we must get into the 2 MW_t/m^3 range if fusion reactors are to become economically competitive.

VI. Impact of Fusion on the Resources of Scarce Materials

Because of the low engineering power density of fusion reactors one must pay considerable attention to the amount and type of elements required for these reactors. Various resources assessments have been made previously(13,37) but here we will take a slightly different approach. In general, we will try to calculate the total installed capacity of fusion reactors required if fusion is to significantly contribute to the energy demand of the 21st century. This very general calculation uses the following assumptions for the period 2000-2100:

1) The average world population is 10 x 10^9 people (it is expected to level out at 12 x 10^9 people in the latter half of the 21st century(81)).
2) The average energy use per capita for the world in this time period will be 5kW_t/cap. (Assuming a 1% per year increase in per capita energy comsumption from the present 2kW_t/cap.)

3) Assume 1/3 of the total energy demand is used to produce electricity (it is now ~28% in the U.S. and projected to reach 50% by the end of the century).[82]

4) Assume an average 40% conversion efficiency to electricity.

5) Assume Fusion will generate 1/2 of the electrical energy in the 21st century.

6) Assume a 30 year plant life.

7) Assume no recycle of radioactive material in the first 50 years after shutdown.

The above assumptions mean that over the next century ~9000 GWe of fusion-electric plants must be installed. Finally,

8) We can use only 10% of the World's reserves for Fusion (assume that future discoveries between now and the introduction of fusion will just balance consumption).

The above set of numbers coupled with the known World's reserves gives the allowable materials investment as listed in Table 4. (Note for example that current designs require 3 to 12 tonnes/MWe of materials that will become activated.)

We see that large amounts of Fe (2700 tonnes MWe) could be used as well as reasonable amounts of Mn (60), Al (20), Cu (4), Cr (4), Ti (2), and Pb (2). The allowable amounts for the refractory metals is rather low [V (0.3), Nb (0.08), and Mo (0.2)]. Nickel represents a problem at 0.3 tonnes/MWe as does Boron (0.7), and the allowable amount of Be is only 0.6 kg/MW_e, or 0.6 tonnes/GWe!

Table 4

SUMMARY OF NON FUEL RESOURCES FOR FUSION

ELEMENT	WORLD RESERVES 10^6 TONNE	ALLOWABLE FOR FUSION-2000-2100 TONNE/MWe
Fe	246,000	2700
Cr	370	4
Ni	24	0.3
Mn	5,500	60
Nb	7	0.08
Cu	390	4
Al	2000	20
Mo	18	0.2
V	26	0.3
Ti	147	2
Be	0.05	0.0006
Pb	160	2
B	66	0.7

We have examined several reactor designs[6,10,13,37,38,80,83,84,85] and compared the average amount of elements required with those allowable in figure 11. Several important conclusions can be drawn.

1) Reactor designs which use Fe, Mn, Cr and Al appear to fit into the previous set of assumptions without too much difficulty.

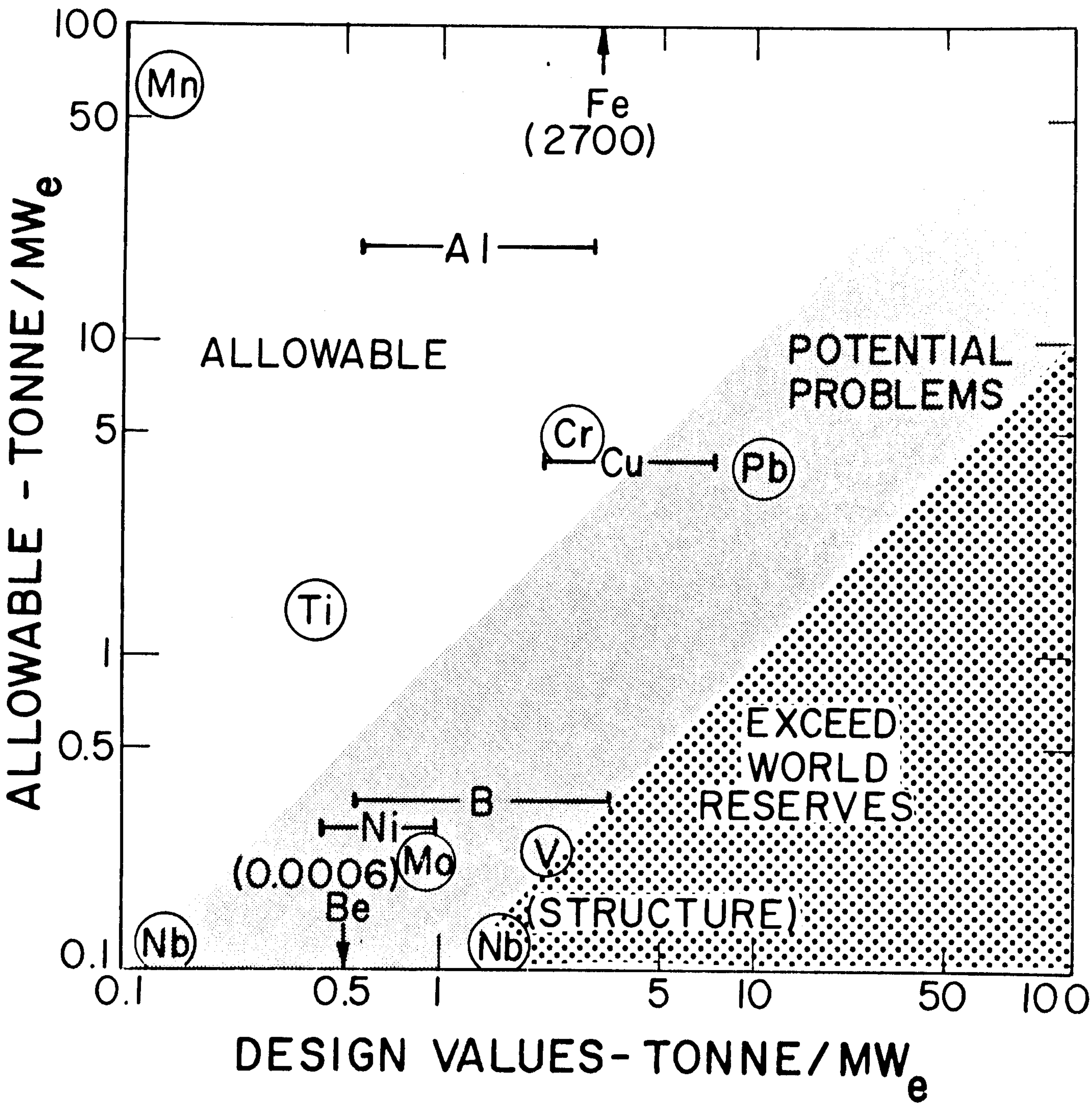
2) The use of Cu, Pb, B, Ni, Mo, and Nb for magnets only would require more than 10% of the present world reserves for the present designs.

3) The use of V, Nb, and Be in the presently anticipated amounts would greatly exceed the world reserves.

Of course, one might argue that if we were willing to pay more for the materials, the reserve picture would be expanded somewhat. However, then one would have to include that in the cost of the plant and factors of 2 increase in the overall fabricated cost of the nuclear island components may be very hard to counter with increased revenues. Nevertheless, these choices should be studied before extensive use of elements in the second and third categories is proposed.

FIG. II

NON FUEL RESOURCE LIMITATIONS FOR FUSION



VII. Summary and Conclusions

We have seen that a great deal of progress has been made in the fusion reactor materials area since 1974 with respect to problem definition, experimental measurements and facility design, and new ideas to extend the useful first wall life. However, there still is a very long way to go before fusion reactor designers can even begin to design high powered (~hundreds of MWe) demonstration reactors and an even longer way to the commercial reactors.

One disturbing trend is the recent tendency to design compact Tokamak reactors (and others as well) with peak neutron wall loadings of 5-6 MW/m^2. The removal of divertors from such systems means that the heat load to the first wall is $\gtrsim$100 $watts/cm^2$, a value much too high for stainless steel first walls.[7,8,30] On the other hand, the present emphasis in the fusion reactor structural materials field is on the austentic steels to be used for at least the next 20-25 years. It is important for both designers and materials engineers to recognize the fundamental thermally induced stress limits in this alloy system and realize that the most economical system (in mills per kilowatt hour generated) may <u>not</u> be that with the highest wall loading!

Finally, much of the current thinking in the fusion reactor materials field in greatly influenced by our experiences in the LWRs and LMFBRs. This has allowed us to make some preliminary estimates of the magnitude and nature of the problems to be faced and in general, this experience has had a positive effect. But we must be careful not to unnecessarily constrain the design of fusion reactors with concepts developed in the fission reactor field. The D-T fusion process does have some very unique properties that can be utilized in innovative and "unusual" ways. The use of ISSEC's (both solid and liquid), solid coolants and unconventional nuclear materials (e.g., Ti alloys) are examples of the flexibility that can be utilized in fusion power reactors. It is important that along with the future progress that will be made in the testing of more conventional materials, scientists keep an open mind to the uniqueness of the energy source we are dealing with and take advantage of the options that have been provided by the fusion process.

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