Fusion Materials - Adapting to Realistic Reactor Environments

G.L. Kulcinski

October 1981

UWFDM-437

Presented at 3rd Technical Committee and Workshop on Fusion Reactor Design and Technology, Tokyo, Japan, October 5-16, 1981.

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DOE/ET52048-21
UC-20

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Abstract

There has been considerable movement in the fusion materials field since the last International Workshop on Reactor Design was held in Madison (1977). Some of the movement has been forward; e.g., we now have much better theoretical descriptions of the melting, vaporization, and electromechanical stresses imposed on first wall material during plasma disruptions. Some of the movement has been "sideways", e.g., the Fusion Materials Irradiation Test Facility (FMIT) now has slipped 3 years in its schedule since the last conference 4 years ago. Finally, there has been very little progress in the fields of pulsed damage simulation for Inertial Confinement Fusion (ICF) Systems, definition of the radiation spectra from ICF targets, or the experimental determination of disruption characteristics for magnetic fusion devices. Several new ideas have appeared since 1977 such as the use of low swelling martensitic alloys, the use of a much more favorable breeding material, $Pb_{83}Li_{17}$, and there have been two major efforts to design materials test facilities: INTOR and TASKA.

1. Introduction

Scientists conducting the first fusion reactor study in 1954 [1] recognized that finding suitable materials for fusion devices was going to be a formidable challenge. However, before Lyman Spitzer and his colleagues could get very far into that area of research, several potentially devastating plasma instabilities were discovered and it took the rest of the 1950's and most of the 1960's to solve them. With the advent of the tokamak in the late 60's and the declassification of some of the inertial confinement schemes in the early 1970's, attention was again directed toward the search for long lasting fusion materials.

The reactor designs of the late 60's and early 70's tended to favor very high neutron wall loadings (in some cases up to 10 MW/m^2) and very high structural temperatures (at times up to 1000°C). Such conditions could only be met by using refractory metals such as Nb or Mo alloys. However, it was soon discovered that it would be difficult to generate, in a reasonable length of time, the industry, the industrial standards, and the irradiation data required to fully qualify refractory metals for a full-fledged fusion reactor economy.

The transition to more conventional alloys, such as austenitic steels, was evident by the time the first Workshop on Fusion Reactor Design was held in Culham, England in 1974 [2]. This transition was relatively complete by the time that the Second Workshop on Fusion Reactor Design was held in Madison, Wisconsin, USA in 1977 [3]. Unfortunately, the austenitic alloys are subject to severe helium embrittlement and corrosion by liquid Li above ~ 500°C and their useful lifetime was thought to be limited to a few MW-years per m^2 even at 500°C. The solution proposed in the 1977 time period was to lower the structural temperature below 500°C, but that in turn caused the overall thermal efficiency to drop below that achieved in LMFBR's.

Other major topics of discussion at the 1977 meeting were [4]: the use of solid breeder materials such as Li_2O, the use of carbon shields to protect the first wall, and the use of liquid metal jets or low pressure gaseous environments to absorb the target X-rays and debris before they could cause melting of the first walls in ICF systems. All of those studies have continued in the 1977-1981 time period.

The object of the present paper is to briefly review the progress (or in some cases, the lack of it) made since the last IAEA sponsored workshop in Madison. The analysis in this paper is broken up into three major areas (see Table I):

A. Metallic Structures.
B. Nonmetals and Coolants.
C. Fusion Reactor Environments.

These three areas are further subdivided into three levels of progress:

1. Considerable Progress.
2. Little Progress.
3. New Ideas (Since 1977).

The reader will certainly recognize that it is not possible in this short paper to cover each of these 9 permutations in great detail. Therefore, we will only briefly discuss a few of the more outstanding examples listed in Table I and leave the reader to follow his main area of interest in the references.

2. Metallic Structures

There have been at least 4 major advances in this area since 1977 (see Table I). Each has had a major impact in the field and it is worth noting that advances in three of the areas are largely attributable to the INTOR [5] study conducted during the 1979-81 time period.

2.1. Considerable Progress

2.1.1. Thermal Response of First Wall Materials During Disruptions in Tokamaks

The deposition of large amounts of plasma energy (100's of MJ) onto the limited surface areas of tokamaks in millisecond time durations will cause most metals to melt and in some cases, vaporize. While this general fact is widely known, it

TABLE I. STATUS OF FUSION MATERIAL RESEARCH - 1981 VS. 1977

	METALLIC STRUCTURES	NONMETALS + COOLANTS	REACTOR ENVIRONMENT
CONSIDERABLE PROGRESS	• FW VAPORIZATION/MELTING • FW LIFETIME ANALYSIS • DIVERTOR PLATE, LIMITER DESIGN • LOW TEMP., HIGH He NEUTRON DATA	• CHEMICAL SPUTTERING - C • HIGH HEAT FLUX COATINGS	• FMIT CONSTRUCTION STARTED • ICF CAVITY GAS RERADIATION • CHARGE X-CHANGE SPUTTERING
LITTLE PROGRESS	• PULSED NEUTRON DAMAGE • "PERMANENT" FW	• LOW T_2 INV.-SOLID BREEDERS • HIGH TEMP. N DAMAGE - C	• PULSED N SOURCE • ICF TARGET SPECTRA • DISRUPTION CHARACTERISTICS
NEW IDEAS (SINCE 1977)	• MARTENSITIC ALLOYS • RAPIDLY COOLED ALLOYS	• $Pb_{83}Li_{17}$ BREEDER/COOLANT • SiC INPORT UNITS	• INTOR • TASKA • "STEADY STATE" TOKAMAK

was largely ignored in the major tokamak studies of the 1970's. The main reason for this omission was the lack of knowledge on the frequency of disruptions and the area over which the energy is deposited. However, during the INTOR project it was felt that a credible design of a near term device (i.e., one to operate in the early 1990's) must consider this phenomenon. Attempts were made to estimate the frequency of disruptions (which have ranged from one shot in 100 to one shot in 1000) and the energy flux to the first wall (which have ranged from a low of $\sim$ 300 J/cm^2 to values over 1000 J/cm^2). Without commenting on the degree of accuracy of these very crude estimates, it is possible to parameterize the problem. Using type 316 stainless steel as a representative material, one can calculate both the melt layer thickness and the amount of material vaporized. A recent model developed by Hassanein et al. [6,7] has improved upon earlier models developed by Behrisch [8], Loebel and Wolfer [9], and Merrill [10]. The Hassanein model [6,7] includes not only thermally varying physical properties and the latent heats of melting and vaporization, but it also includes vapor shielding of the metallic surface and redeposition of evaporated material on the original surface. The melt layer thickness for steel is shown in FIG. 1(a) as a function of the energy flux and for disruption times between 5 and 20 milliseconds. This figure shows that melt layer thicknesses of 100-200 microns can result from energy fluxes of 300-800 J/cm^2. The shorter the time duration the thinner the melt region. Such a paradox occurs because more of the disruption energy goes into vaporization during the shorter pulses. Erosion due to vaporization ranges from 1 to 100 microns per shot depending on the disruption time and whether or not the vapor shielding is included (see FIG. 1(b)). More detailed analyses of steel, carbon, and Mo appear elsewhere [11], but it is now clear that if the melt layer in steel is not stable in the rapidly changing magnetic fields associated with a disruption, then those fusion devices which cannot stand more than 1 cm erosion per 1000 disruptions are limited to less than 180 J/cm^2 in 20 ms or 100 J/cm^2 in 5 msec (see FIG. 2). As a final comment, it is seen that for INTOR level energy fluxes (300-600 J/cm^2) the first wall may be eroded away in 30 to 100 disruptions.

2.1.2. First Wall Life Analyses

The analysis of first wall lifetimes, especially in tokamaks, has become much more quantitative than was evident in the 1977 meeting. In particular, two types of fatigue lifetime analyses have been performed. The S-N analysis, based on maximum surface stresses, has been applied to the INTOR project [11]. Another type of analysis considers the thermal stresses generated by a temperature gradient, the growth rate of detectable cracks under the influence of those stresses, and the reduction of the stresses as the thickness of the wall is reduced [12]. An example of the first type of analysis for INTOR is shown in FIG. 3. It reveals that the allowable thickness of

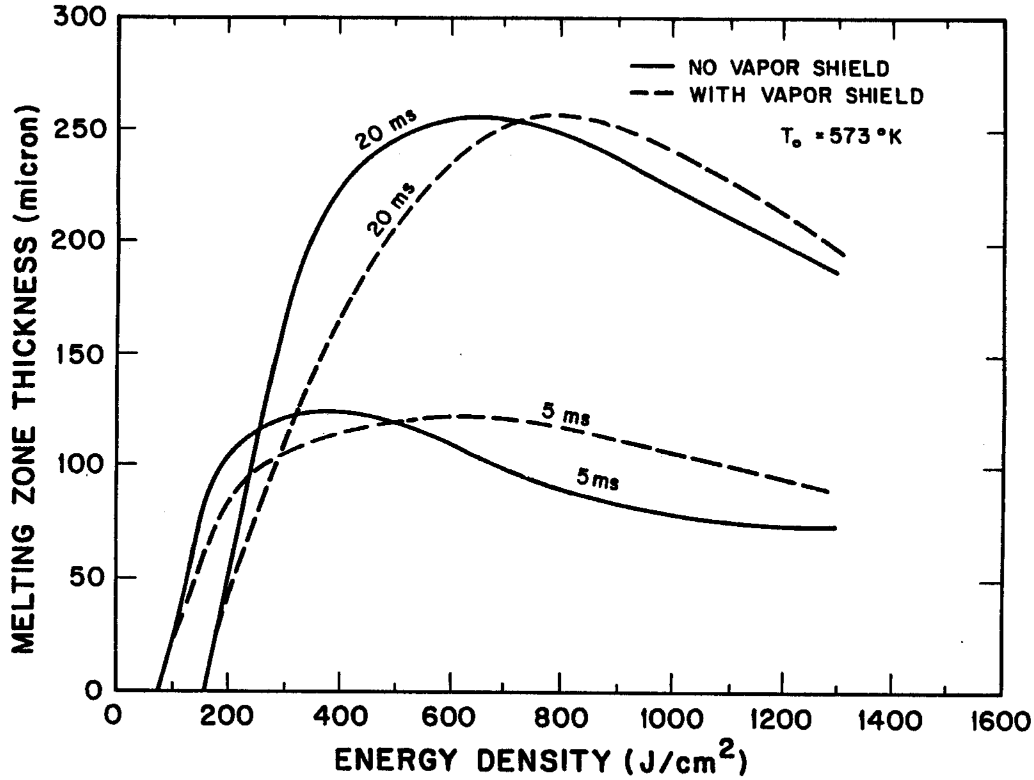

FIG. 1a. Effect of vapor shielding and disruption time on melt layer thickness of stainless steel [7].

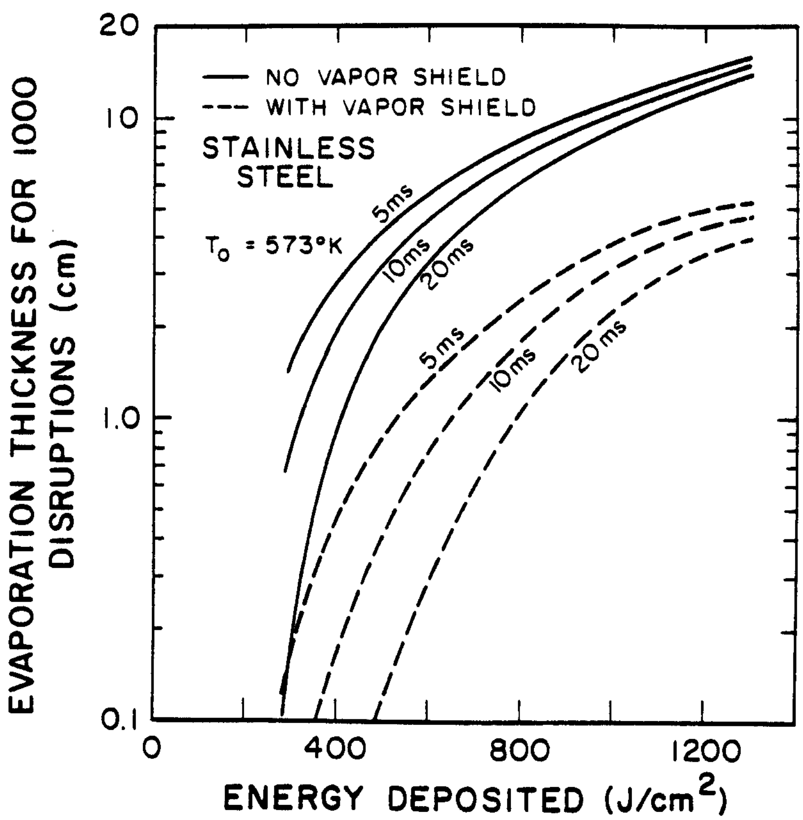

FIG. 1b. Effect of vapor shielding and disruption time on evaporation of stainless steel [6].

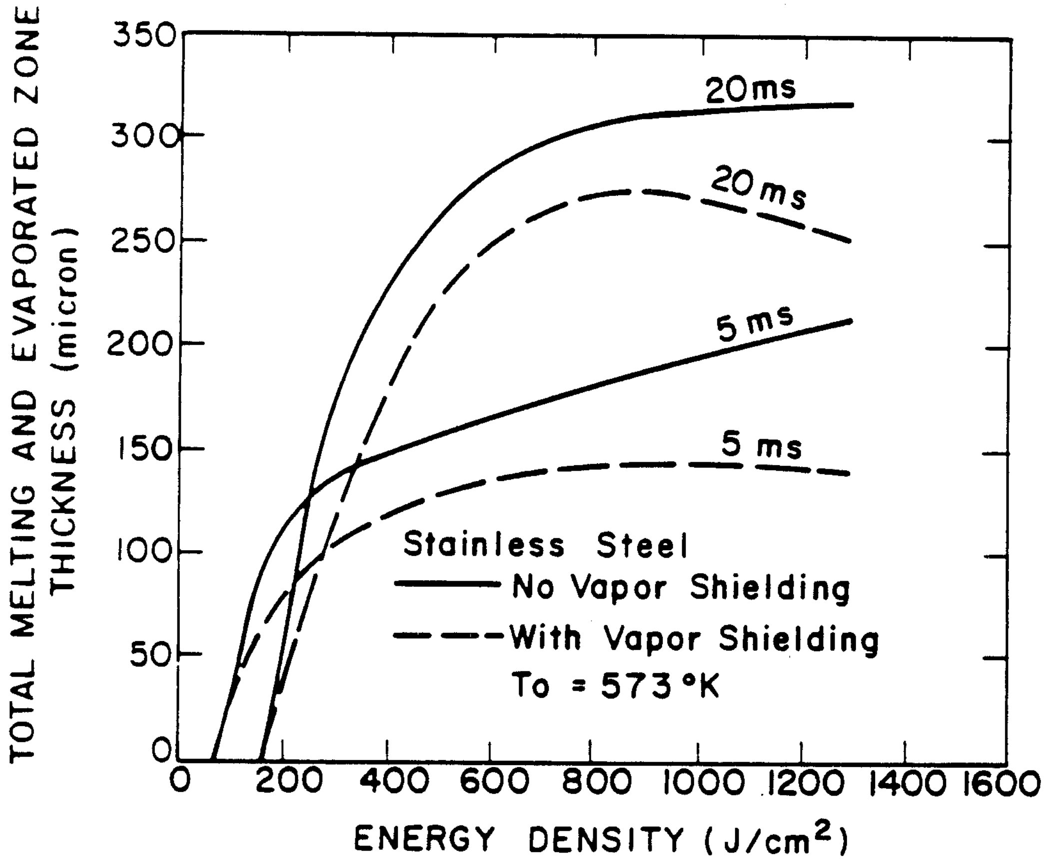

FIG. 2. Effect of vapor shielding and disruption time on the total thickness of stainless steel effected by high heat flux [7].

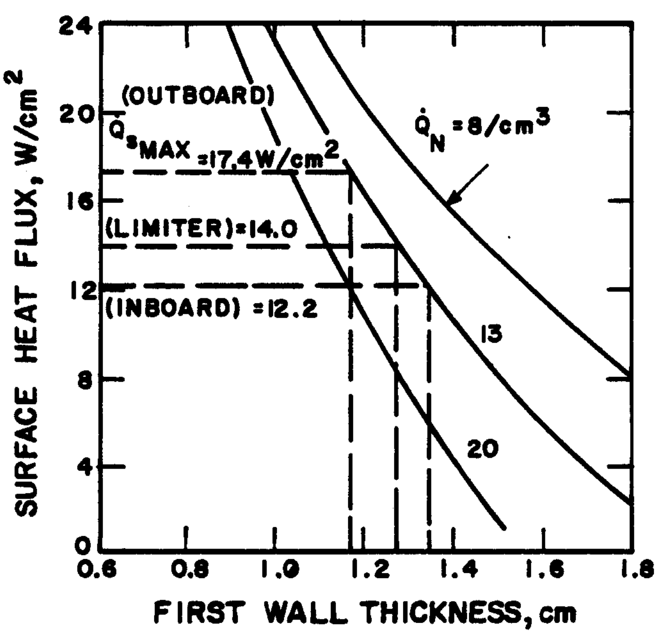

FIG. 3. Allowable maximum surface heat flux in a Type 316 stainless steel wall for a fatigue life of 7.1×10^5 cycles, as a function of wall thickness and nuclear heating rate [12].

an INTOR first wall is quite sensitive to the heat flux and the neutron heating rate [11]. The main message of FIG. 3 is that for relevant tokamak heat fluxes of 10-30 watts per cm^2, the allowable first wall thickness ranges from 1.4 to 0.8 cm. This thickness also has to be sufficient to withstand erosion due to charge exchange neutrals (see section 4.1), one of the main problems of tokamak reactors.

2.1.3. Divertor Plate Design

The design of divertor collector plates for tokamaks has been a difficult and essentially unsolved problem even since the publication of the first large scale tokamak designs using divertors: UWMAK-I [13] and the PPPL reference reactor [14]. The basic problem is that the collector plate has to withstand high heat fluxes (~ 1 to 2 kW/cm^2) and high erosion rates (~ several cm/year) due to the D, T, and He ions. Past designs have used liquid films [13], showers of liquid metal [15], and high velocity H_2O cooled refractory metal tubes [16]. Another solution, recently proposed by scientists working on the INTOR project [17], was to use tungsten tiles, attached to water cooled steel blocks. The high heat flux is reduced by inclining the W tiles to the incoming beam. The tiles, which reach ~ 2000°C, radiate the energy to water cooled walls. The sputtering erosion rate is 2 cm per full power year which requires replacement every 275 full power days. Hence, even after 10 years of research on this topic, we still do not have a solution which will work for more than 1 FPY of commercial reactor operation.

2.1.4. Limiter Design

Another approach to the impurity problem that has been pursued in the last few years is the pumped limiter. Examples of this approach include NUWMAK [18], STARFIRE [19], and the US-FED [20]. These limiters, such as that shown in FIG. 4, basically "skim" off a small fraction of the plasma and the impurities. The main problem is that both the heat flux and sputtering associated with the limiter are very severe. For example, if the plasma strikes the limiter in FIG. 4 uniformly, the heat flux to the limiter ranges from 400 W/cm^2 at the tip to ~ 200 W/cm^2 on the flat portion of the limiter. While such heat fluxes can be handled with appropriate design, they do require rather thin (1.5 mm) members to avoid excessive high temperature and stresses. Unfortunately, physical sputtering and erosion require thick members to achieve long lifetimes. The combination of these two factors in STARFIRE produce a limiter wall thickness of 0.15 cm. Therefore, as in the case with divertors, present design solutions will probably have uncomfortably short lifetimes. While there has been considerable progress in this area, a lot more work needs to be done.

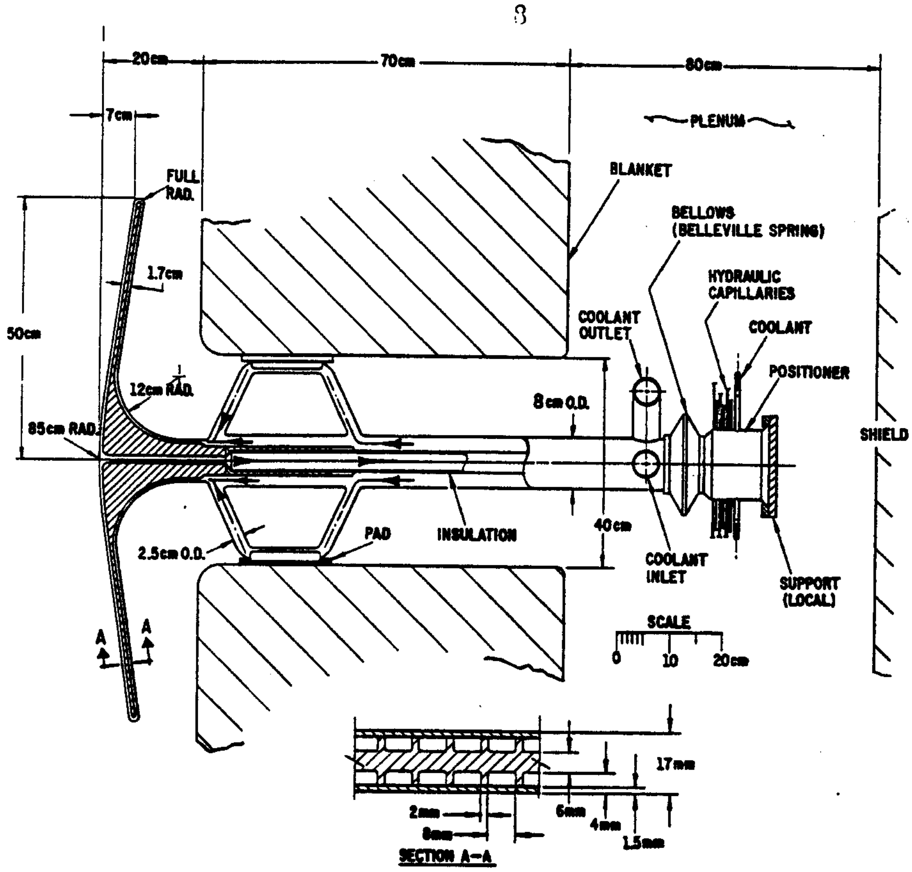

FIG. 4. Cross section of the STARFIRE limiter design [19].

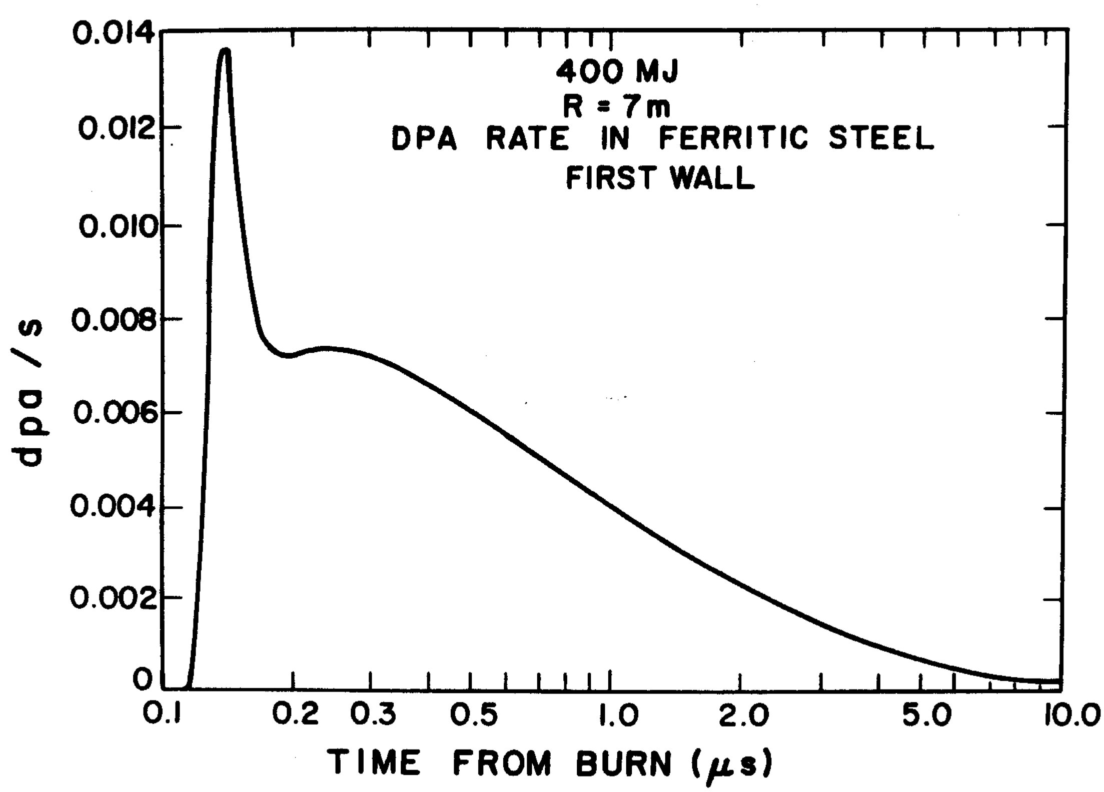

FIG. 5. Damage rate in HIBALL reactor metallic walls [34].

2.1.5. Low Irradiation Temperature Data

Another major area of progress since the last meeting is the data obtained on low temperature (55°C) neutron irradiation of steels in thermal reactors where high levels of helium are produced [21]. The testing of 20% cold worked 316 SS at 35°C has shown that after displacement damage levels equivalent to ~ 1 MW-y/m^2 and helium contents equivalent to 1 to 2 MW-y/m^2, the uniform ductility is reduced to < 0.6%. However, the total elongation at failure is still in the 10 to 15% range. This latter data indicates that fusion reactor walls, operated at low temperatures, may be able to produce some plastic deformation to relieve stresses which temporarily exceed the yield strength. On the other hand, the low uniform elongation values indicate that under conditions where constant loads produce stresses above the yield strength, brittle failures may occur quite frequently.

2.2. Areas of Little Progress

2.2.1. Pulsed Damage in Metals

It is now clearly established [22-23] that the metallic structural components of a DT-ICF reactor will see very high instantaneous damage rates from neutrons and target debris (see FIG. 5). It is also well known that some metallic components may also experience high heat fluxes from X-rays and target debris [24-28]. However, what is not known is how these materials will react to such high damage rates. Some theoretical analysis has been attempted by Ghoniem [29-30] but the level of effort needs to be increased and experimental information with pulsed neutron or ion sources is sorely needed (see section 4.2.1).

2.2.2. Permanent First Wall

Since the Madison meeting there have been no full scale ICF reactor designs published and only three full scale conceptual magnetic fusion reactor designs reported: NUWMAK [18], WITAMIR-I [31], and STARFIRE [19]. The wall lifetime estimated for each reactor was 2 FPY for the Ti alloy in NUWMAK, 4 FPY for the ferritic steel in WITAMIR-I, and 6 FPY for the unspecified alloy in STARFIRE. While each of these estimated first wall lifetimes is subject to a considerable amount of uncertainty, it is clear that we are still a long way away from permanent first walls in commercial magnetic fusion reactor designs.

2.3. New Ideas

2.3.1. Martensitic Alloys

Spurred on by success in the fission reactor field, materials scientists have recently turned to the use of martensitic (or sometimes referred to as ferritic) steels in the neutron environment of a fusion reactor [32]. One of the first reasons for this trend is the greater resistance of the ferritic steels to void formation [33] (see FIG. 6). This, coupled with higher thermal conductivity of the ferritics, lower Cr and Ni contents, and better compatibility with lithium has caused scientists to examine the use of HT-9 in the WITAMIR-I [31], HIBALL [34], and TASKA [35] designs. Disadvantages of this alloy include a substantial DBTT shift if the steel is irradiated below 300°C, and its ferromagnetism. The problem with ferromagnetism is not so much with respect to field perturbations in the plasma but in the forces on metallic components that enter and leave the plasma region. Attaya [36] has recently developed codes to examine these effects in the tandem mirror test facility TASKA [37] and found that the loads on the HT-9 coolant pipes into and out of the central cell magnetic field could be handled by conventional design procedures.

2.3.2. Rapidly Solidified Metals

This class of alloys has been recently examined mainly by scientists at MIT [38]. The main advantages of such rapidly solidified alloys are expected to be: (1) the flexibility in the composition of the alloy, (2) the homogeneity of the alloys (i.e. lack of large incoherent precipitates), (3) high strength, and (4) resistance to radiation damage. High strength, high thermal conductivity copper alloys are currently being investigated [39].

3. Nonmetals and Coolants

3.1. Considerable Progress

3.1.1. Chemical Sputtering of Carbon

The use of carbon in fusion reactors as a structural material was proposed by scientists at GA [40] in the early 1970's and in 1974 proposals to use carbon cloth [41] or thick carbon layers (ISSECS) to protect the wall from neutron damage were made [42]. Later, carbide coatings [43] and even limiters made from carbon were considered. By the late 70's, carbon limiters were actually used in D-III [44] and PLT [45]. One of the major concerns about the use of carbon near the plasma of a DT fusion device was the so-called "chemical sputtering" in which the impinging hydrogen would react with the carbon to form methane or acetylene. The main progress in this area has been the discovery that chemical sputtering may not be as bad as originally thought. Experimental evidence from the Japanese [46] has shown that chemical sputtering coefficients with hydrogen are only in the 0.01 range, not much different than the physical sputtering values. Recent calculations by Smith [47] (see FIG. 7) show that the CH_4 and C_2H_2 formation is not more than physical sputtering alone for the normal carbon operating temperatures. Hence the erosion rates will not be

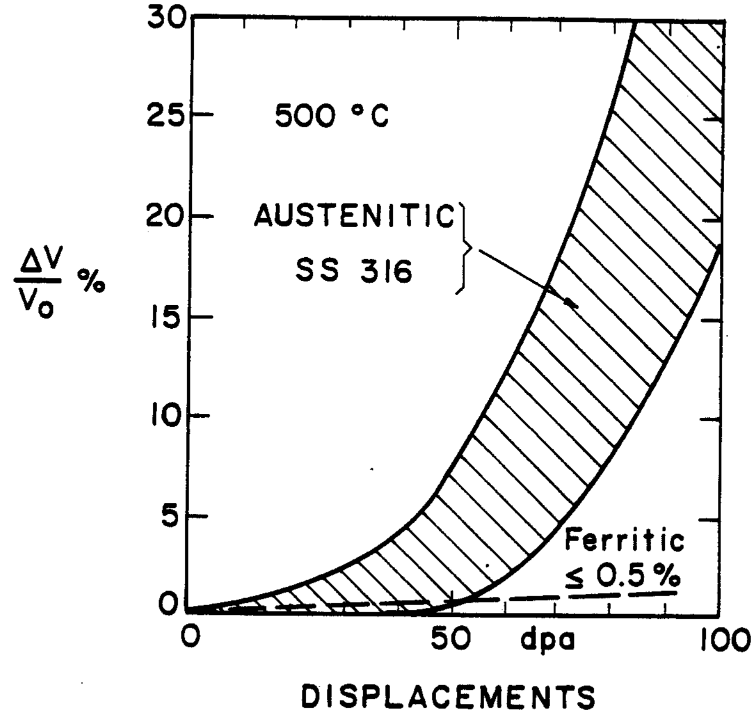

FIG. 6. Swelling behavior of austenitic and ferritic steels irradiated in EBR-II.

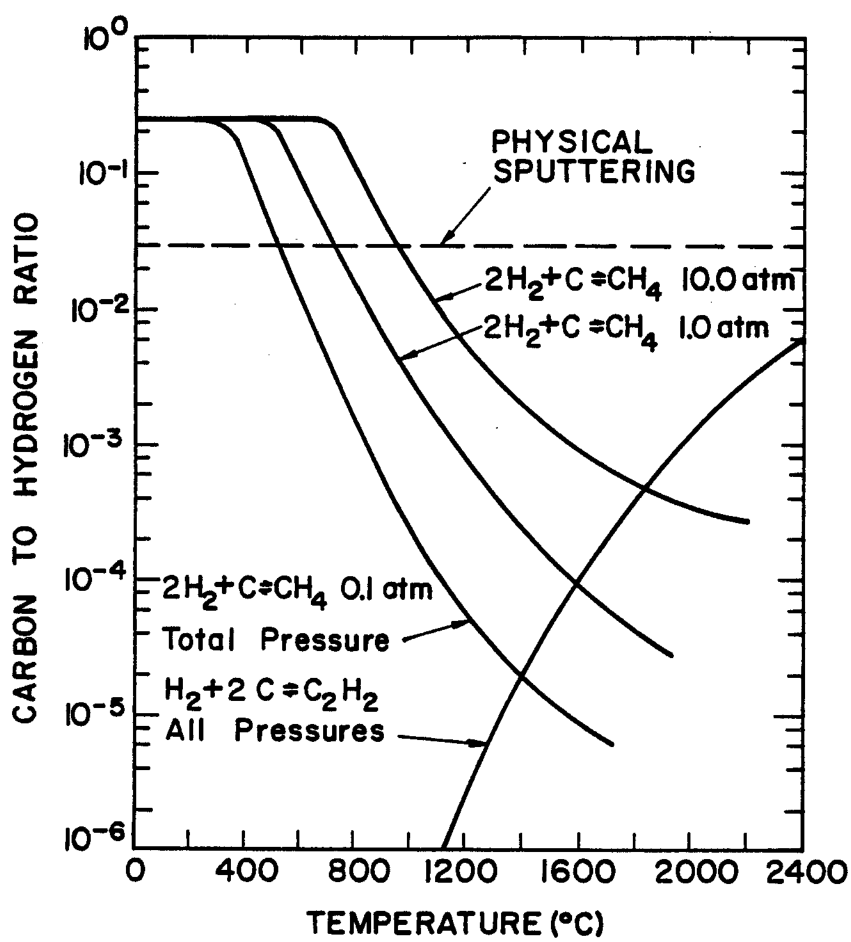

FIG. 7. Calculated sputtering erosion rates of carbon based on thermodynamic equilibrium data [47].

abnormally high, as was originally thought, nor should the contamination of the plasma be particularly severe. This information is particularly important because it means that scientists can now utilize the high temperature capabilities of carbon in regions of high heat flux, i.e. beam dumps, limiters, divertor channels, armor against disruptions, etc.

3.1.2. High Heat Flux Protective Coatings

Considerable progress has been made in the development of coatings that can withstand extremely high heat fluxes. For example, scientists at Sandia Laboratory have developed TiC coatings to cover graphite which can stand heat fluxes up to 1 or 2 kW/cm^2 [48]. These coatings have remained functional for over 2000 five-second shots and can be used in beam dumps, limiters or other areas where high heat fluxes are anticipated.

3.2. Little Progress

3.2.1 Low Tritium Inventory Solid Breeders

There was a great deal of excitement about the use of solid breeder materials at the 1977 conference [4]. The advantages of these materials are a lack of chemical reactivity and, in the case of Li_2O, a higher Li atom density than pure lithium itself. Originally it was thought that the tritium inventory would be low because the tritium diffusivity in these solid breeders may be high. However, it was subsequently discovered [49] that gas phase mass transport was as important, or even more important, as the diffusivity. Another factor in the inventory will be the amount of tritium trapped by irradiation produced defects. The magnitude of this trapping is unknown at this time but an experiment, TRIO-1, is currently being planned to measure such effects [50]. An illustration of the uncertainty associated with the estimate of tritium inventory in the solid breeder materials can be obtained from the recent STARFIRE [19] design. It was estimated that if the inventory is governed by thermal diffusivity alone, there would be ~ 0.1 kg of tritium in the 3000 MW_{th} plant. However, if gas phase transport and radiation trapping are important, the inventory could be as high as 300 kg! Clearly the latter number is unacceptable and indicates that much more work is needed if we are to achieve low T_2 inventory solid breeders.

3.2.2. High Temperature Neutron Damage in Graphite

There has been essentially no new information about the effects of neutron damage on carbon, or graphite, since the 1977 meeting. This is particularly discouraging since there are a large number of proposed uses for carbon in both ICF and magnetic fusion reactors. The lack of such studies is evident in all of the major fusion programs around the world.

3.3. New Ideas

3.3.1. $Pb_{83}Li_{17}$ Breeder Coolant

Perhaps the most important new idea in breeder/coolants has been the proposed use of $Pb_{83}Li_{17}$ eutectic alloy for a coolant/breeder in fusion reactors [51]. This alloy, which melts at 234°C, has several attractive features, some of which are listed below.

1. Low chemical reactivity with water even at 500°C [52].
2. High neutron multiplication and high tritium breeding ratio.
3. Extremely low tritium inventory and easy tritium extraction.

As an illustration of the last point we quote some recent experimental measurements made by Veleckis [53] and some theoretical calculations by Ortman [54]. The solubility of T_2 in Pb-Li alloys is shown in FIG. 8. It can be seen that the hydrogen solubility in the $Pb_{83}Li_{17}$ alloy can be easily kept to less than 1 appm in the 500-770°C range. Not only is such a value important from a safety standpoint, but the high partial pressure of hydrogen in $Pb_{83}Li_{17}$ means that it can be easily extracted from the coolant.

3.3.2. SiC INPORT ICF First Wall Protection Schemes

There have been many proposals to protect the first walls of ICF reactors ranging from wetted walls [55], magnetic fields [56], gases [57], and thick sheets of liquid metals [58]. These concepts are summarized in Refs. [59-61]. One of the more attractive schemes is the HYLIFE [62] concept based on liquid Li metal jets. However, one major drawback is the low repetition rate because of the disassembly of the jets after each pulse and the low temperature increase of the Li passing through the chamber (~ 10°C). A solution to this problem has been proposed in the HIBALL [34] project which combines the advantages of many of the previous schemes.

The flow rate of the liquid metal, $Pb_{83}Li_{17}$ in the case of HIBALL, is slowed down by forcing it to flow through flexible, porous SiC tubes (see FIG. 9). These tubes are called INPORT units for <u>In</u>hibited Flow in <u>Por</u>ous <u>T</u>ubes. The INPORT units also prevent the disassembly of the jets after each shot, allowing the energy from ~ 10 to 20 shots to raise the temperature of the liquid by 100 to 200°C during each pass through the chamber. The surface of the tubes is protected from the target X-rays and debris by a film of liquid metal either seeping through the porous structure or from the condensation of $Pb_{83}Li_{17}$ vaporized in the cavity during each shot.

SiC was chosen for the woven structure because of its neutron damage resistance and because of its compatibility with

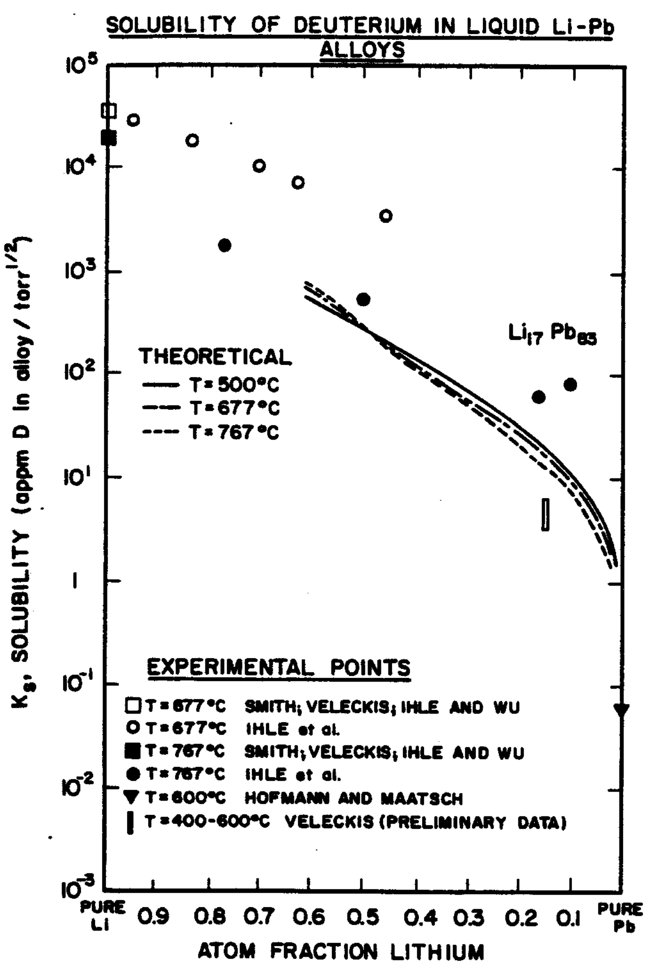

FIG. 8. Solubility of deuterium in liquid Li-Pb alloys.

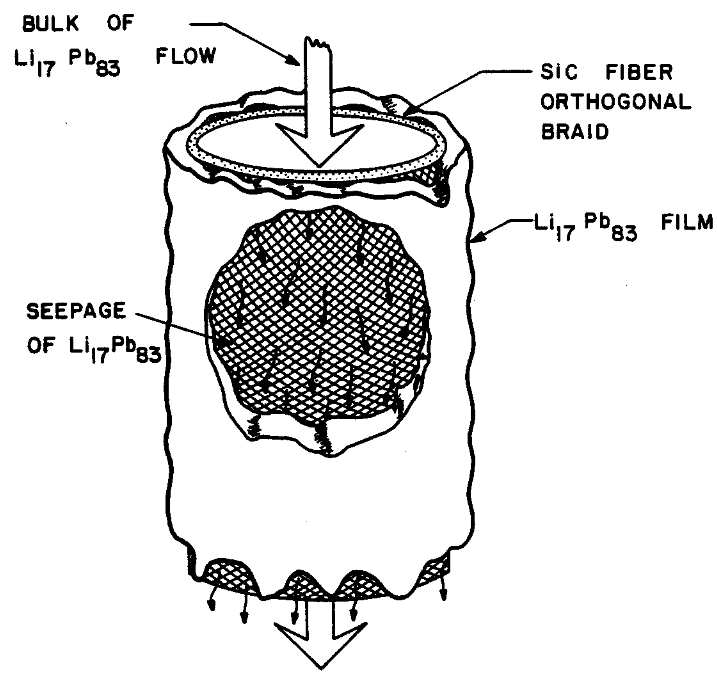

FIG. 9. Schematic of inhibited flow in porous tube unit for HIBALL.

Pb-Li alloys at the HIBALL operating temperature. The use of many layers of INPORT units will greatly moderate the energy of the neutrons and thereby reduce the radiation damage to the first metallic walls [63]. Finally, the INPORT units can be easily replaced; they are relatively inexpensive, and the residual radioactivity of the tubes is very, very low.

4. Reactor Environment

4.1. Considerable Progress

4.1.1. FMIT Design

Four years ago there were three main fusion neutron test facilities being constructed or planned. Since that time one has been constructed (RTNS-II), one has been cancelled (INS) and the remaining facility (FMIT) has run into delays which have moved its startup date back by at least 3 years. It is clear that the FMIT design is much better developed than it was in 1977 and in fact, construction on the device has begun. However, funding for the completion of the device has been sharply cut back because of budgetary difficulties in the U.S. and the testing flexibility of the device has been reduced by removing one of the two test cells. It is now anticipated that it will go into operation in late 1986.

4.1.2. Cavity Gas Reradiation Characteristics

The current concept of Light Ion Beam (LIB) fusion reactors rely on relatively high gas pressures (~ 50 to 100 torr) to assist the propagation of the light ion beam to the target. This gas density is also high enough to absorb the target X-rays and debris before they hit the wall. However, in the process of absorbing the target debris, a phenomenon similar to a "fireball" is developed. The energy absorbed by the gas alone is subsequently reradiated, at long wavelengths, to the chamber walls. The reradiation time is quite important and is a function of the gaseous element plus any impurities present. Obviously, the longer the reradiation time the lower the first wall temperature rise. Since 1977, models have been developed to describe the reradiation phenomenon in ICF reactors [28,64] and these led to the suggestion that small amounts of alkali metals could significantly increase the reradiation time and/or reduce the energy that reaches the first wall. FIG. 10 shows that as little as 0.3% of Na impurites in argon gas can reduce the energy reaching the wall by a factor of 3 and the maximum heat flux by a factor of 10. The price to be paid is a 50% increase in overpressure to the first wall.

4.1.3. Charge Exchange Neutral Sputtering

Sputtering of the first walls by ions escaping the plasma has always been of concern to the materials scientist. In the past, most of the attention has been paid to the charged particles which directly strike the first wall but recent calculations from the INTOR project [5] revealed that sputtering from charge exchange neutrals could also be quite important. These neutral particles are thought to be in the 100-200 eV range which is, unfortunately, at the maximum of the sputtering curve for most metals. Furthermore the flux of such particles in a real device is only known to an order of magnitude at best. For INTOR, the flux of neutral D, T, and He is 1.5×10^{16}, 1.5×10^{16}, and 1.7×10^{15} $cm^{-2}s^{-1}$, respectively, and this causes a maximum erosion of 2.3 mm of 316 SS per full power year [17]. The actual erosion rate could even be much higher because of the large uncertainties in the particle flux around beam ports, divertor collector plates, limiters, etc. Future work is needed to come up with more precise values of these charge exchange fluxes.

4.2. Not Much Progress

4.2.1. Pulsed Neutron Source

As was pointed out in section 2.2.1, we now have a reasonably good idea of the neutron spectra associated with ICF targets. We know that the damage rate is 0.005 to 0.01 dpa/sec for approximately one microsecond and the few attempts which have been made to model this damage implied that phenomena such as void swelling may be less in ICF systems than in a steady state damage environment [29,30]. However, there are no facilities presently available, or even planned to be able to test these hypotheses at even modest damage levels. Single shot test facilities such as those proposed by Sandia and Wisconsin scientists [65] will not be able to accumulate sufficient fluence to exceed the incubation threshold for void nucleation. The lack of a suitable ICF test facility is still as much a problem in 1981 as it was in 1977.

4.2.2. Target Spectra Information

A particularly important input to any ICF reactor design is the target X-ray and debris spectra. This is especially true in those concepts where the cavity background pressure is less than 1 torr. Unfortunately, exact X-ray and target spectra are not available because: (1) there is no direct experimental data on reactor relevant targets, and (2) the target designs considered most likely to work are classified. In the 1977 meeting, there was a direct appeal for release of some of the calculated spectra but the situation has not changed in the last 4 years. Therefore, first wall and cavity designers are forced to either choose arbitrary spectra, or to parameterize their designs such that, when and if relevant spectra are available, one could choose an appropriate operating point. Such a situation is highly unsatisfactory and has an obvious retarding effect on progress in this field.

4.2.3. Disruption Characterization

While we have graphic experimental evidence that disruptions frequently occur in present day tokamaks we know very little about:

A. their frequency,
B. their time duration, or
C. the spatial extent over which they deposit their energy.

Recent studies have attempted to estimate these three quantities, but there are so many possible combinations that we can only claim we know the characteristics of a disruption to the first order of magnitude. This can be amply illustrated from the INTOR project. The plasma energy can be calculated, with reasonable accuracy, to be 200-250 MJ. The next question is what fraction of that energy is available for melting and vaporization? Experiments tell us that a disruption is usually accompanied by a burst of X-rays which can carry perhaps 10-30% of the energy uniformly to the first wall. Next, the area over which plasma ions strike the first wall must be known. Experience now indicates that disruptions most often move to the inboard region, comprising ~ 30% of the total first wall area. However, there is no assurance that the plasma will be uniformly deposited over that 30% and it is not inconceivable that peaking factors of 2 to 3 will occur. FIG. 11 illustrates the range of these variables and it is seen that values as low as 290 J/cm^2 to values of 1300 J/cm^2 are possible for INTOR.

Finally, the time period over which the energy is deposited is important as we saw in FIGS. 1 and 2. Time durations as low as 1 ms have been predicted as have values of 20 ms. When all of these variables are combined for steel, it is seen that the thickness of a steel first wall affected by a typical disruption ranges from 100 to 300 microns per disruption. Futhermore, the frequency of disruptions is not known to be better than an order of magnitude, i.e. it could range from 0.1% to 1%. With such a wide range of variables it is obviously not possible to say at this time that <u>any</u> metallic first wall will last the lifetime of <u>any</u> tokamak power reactor now under consideration.

4.3. New Ideas

4.3.1. INTOR - A Tokamak Technology Test Reactor

In 1979, a joint study between the USA, USSR, Japan, and the European community was initiated to design the maximum reasonable step beyond the TFTR, JT-60, JET, and T-15 class of experimental tokamaks. This study produced an extensive data base assessment in mid-1980 [5] and by mid-1981 it had produced a unified design [17]. The details of this device are described elsewhere, but from a materials aspect, it is the second major high volume, high damage level tokamak to be

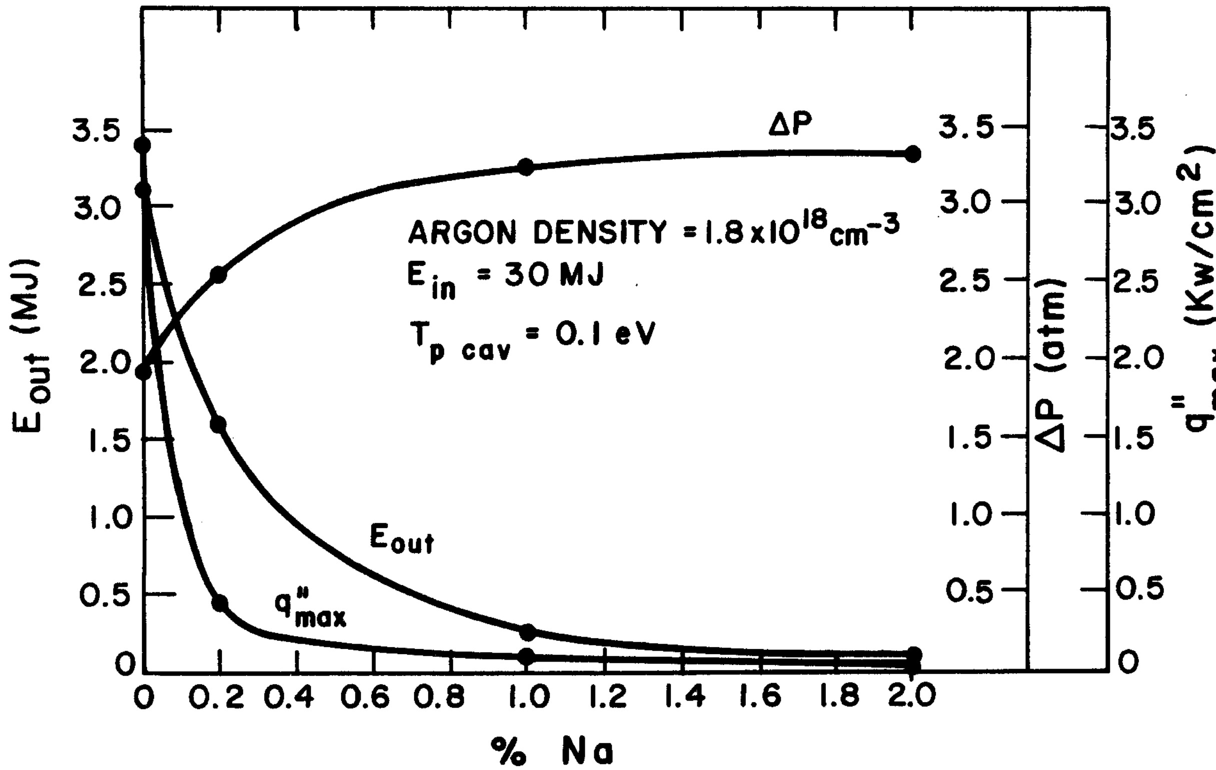

FIG. 10. Response of an argon cavity gas to a 30 MJ pellet explosion measured at a 4 meter radius wall vs. sodium concentration.

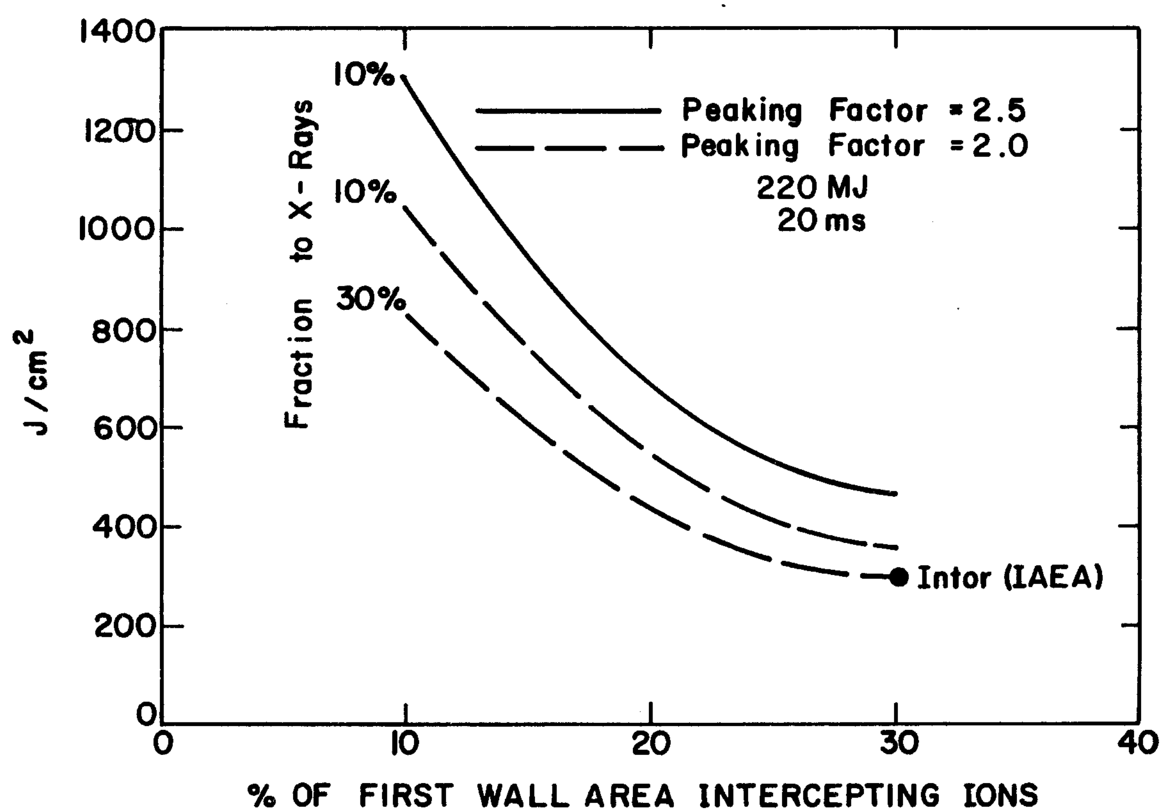

FIG. 11. Effect of deposition area on energy flux from INTOR plasma disruption.

designed (TETR [16] was the first in 1976). The total cost of INTOR (~ 3 billion dollars) represents a formidable investment for any one country. However, if it is shared among many countries, it may be possble to build such a device by the early 1990's.

4.3.2. TASKA - A Tandem Mirror Technology Test Facility

Spurred on by successful physics experiments, and the promise of an economical, maintainable, and simple commercial reactor (e.g., WITAMIR-I [31]), scientists at the University of Wisconsin and Karlsruhe Nuclear Laboratory in the FRG initiated a Tandem Mirror Technology Test Facility Study [35] with goals similar to INTOR. However, the TASKA device delivers even higher damage levels (1.5 MW/m^2 vs. 1.3 MW/m^2 for INTOR) at lower DT fusion power levels (86 MW for TASKA vs. 620 MW for INTOR). When one considers the increased duty cycle of TASKA (steady state versus 0.7 to 0.8 for INTOR) and the lack of thick first walls to withstand high heat fluxes, the rate of damage accumulation is much higher for TASKA (see FIG. 12). The details of this facility are given in Refs. [35,37].

4.3.3. "Steady State" Tokamaks

It has long been recognized that the pulsed nature of tokamaks is a serious disadvantage which needs to be overcome. In an attempt to remove this disadvantage of tokamaks, scientists have been looking for ways to keep the plasma current flowing without the use of magnetic coils. One recent attempt to highlight the advantages of a steady state tokamak over a pulsed system is documented in the STARFIRE design [19]. In this system the current is driven by RF sources. If this concept works, the removal of fatigue loads should be a great advantage to the overall operation of such a system.

5. Conclusion

We have briefly touched on the progress, or lack of it, in important areas of fusion materials research. Generally, one has to be impressed with what we have learned in the past 4 years. One area where it is obvious that more effort is needed is in the area of ICF. The lack of general progress in this field since the Madison meeting is indeed disturbing and every effort to remedy the situation needs to be made. On the positive side, the INTOR project has spurred the largest coordinated effort ever to assess the materials aspects of a tokamak device. This type of effort should also be mounted in the area of tandem mirror and ICF systems.

Acknowledgment

The author would like to thank the Magnetic Fusion Energy Office of the Department of Energy and the Wisconsin Electric Utilities Research Foundation for partial support of this

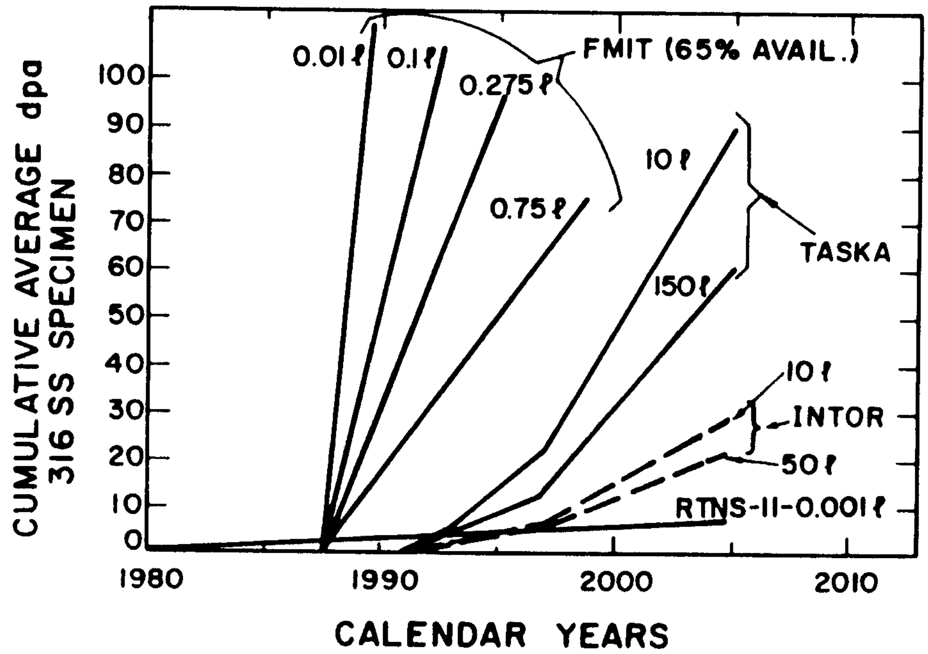

FIG. 12a Cumulative damage in fusion materials test facilities.

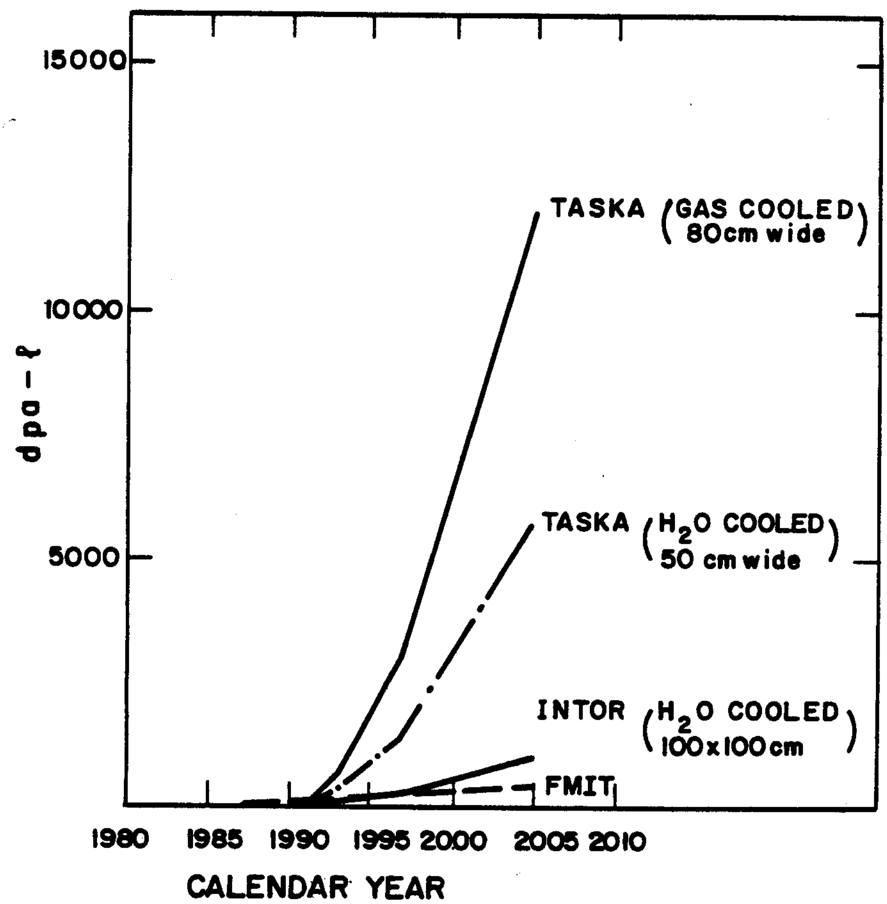

FIG. 12b. Comparison of the damage-volume figure of merit for near term materials test facilities.

project. The author also wishes to thank the many members of the University of Wisconsin Fusion Engineering Program who have contributed ideas and suggestions to this area over the past 10 years.

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