Materials Problems and Possible Solutions for Near Term Tokamak Fusion Reactors

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by

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1. Introduction

As the world-wide fusion community progresses beyond the TFTR/JET/T-20 phase of Tokamak devices,[1] it will begin to encounter the first serious DT neutron damage problems in reactor structures. These problems will first show up in the experimental power reactors (EPR's) which are presently designed to produce some electricity (hence high temperatures) by converting the kinetic energy of the high fluxes of the 14 MeV neutrons and 3.5MeV helium ions to heat and eventually useable power.[2] It has been a common fault to dismiss the neutron damage problems in EPR's as insignificant compared to those in the next generation of Demonstration Power Reactors (DPRs) which are required to produce electricity on a steady-state basis. However, closer examination of the anticipated operating conditions of EPR's reveals that there could be some very serious problems with present day materials.

It is the purpose of this paper to clarify the magnitude of the problems that might arise from neutron damage and to put into perspective the methods and facilities that might be used to solve these problems.[3] First, a brief review of some of the fundamental aspects of radiation damage from neutrons will be given for the non-materials scientist* followed by a current listing of the anticipated radiation environment of the various near term (TFTR, JET, T-20), EPR, and DPR designs. The reader should note that such designs are highly fluid and may change considerably in the future (in fact due to the very problem we will be discussing). Next, the present and future facilities that could be used to test CTR materials will be reviewed and their utility in providing pertinent fundamental and engineering data will be discussed. Finally, some conclusions and recommendations on the near term reactor materials problems will be presented.

2. Background Information on DT Neutron Damage in Potential EPR and DPR Materials

When an energetic neutron strikes any solid material, it produces damage in a variety of ways. It is convenient to think of the damage process as broken up into primary and secondary responses of the material (see Figure 1). The primary responses are the displacement of atoms from their equilibrium sites via elastic, inelastic and nonelastic events, and the transmutation of some elements into different elements, or

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* This chapter may be skipped by those familiar with the radiation effects field.

Figure I
Schematic Of The Response Of Materials To Neutron Damage

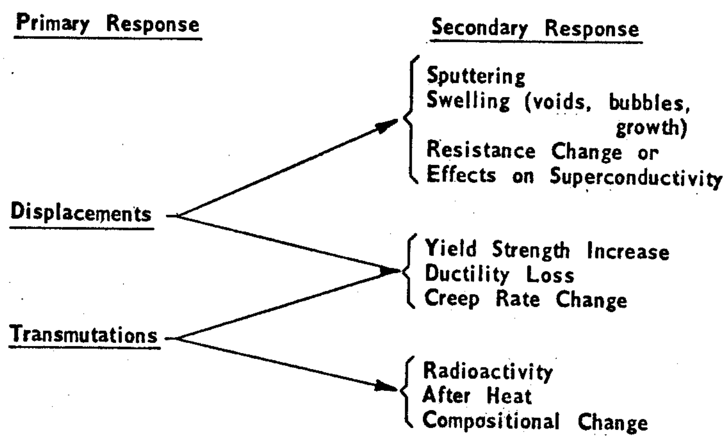


Table I

Typical Displacement and Gas Production Rates in Metals

	Fusion Reactor First Wall 1 MW/m^2			First Fission Test Reactor-EBR-II (max)		
Material	dpa/yr	appm He/yr	appm H/y	dpa/yr	appm He/yr	appm H/yr
SAP[(a)]	17	410	790	76	7.9	50
316SS[(b)]	10	200	540	44	4.7	270
Nb	7	24	79	28	1	6.6
Mo	8	47	95	30	1.8	3.5
V	12	57	100	54	0.5	14
C	10	2700	Neg.	5	130	Neg.
Be[(c)]	(d)	2800	130[(e)]	(d)	3300	Neg.

(a) SAP = Sintered Aluminum Product, 5-10% Al_2O_3 in Al

(b) SS - Stainless Steel

(c) ~ Typical of 5 cm from first wall

(d) Displacement cross section not available

(e) Tritium

isotopes of the same element. These primary interactions can then produce secondary responses in the solid via the migration of the defects to internal sinks, or the formation of microdefects which change the physical properties, dimensions, and mechanical response of the material. Let us turn our attention to the primary interactions first.

2.1 Primary Responses

The cross section for displacing atoms is a rather complex function of neutron energy due to the multitude of reactions that can take place. In general, the displacement cross section increases rapidly with neutron energy up to ~1 MeV and rises somewhat slower after that (see Figure 2). It can be seen from that figure that the displacement cross section at 14 MeV is only 3-4 times the displacement cross section at 1 MeV for high Z elements. For low Z elements the displacement cross section is relatively constant with energy above 1 MeV. This latter effect is due to the anisotropic scattering at high neutron energies and is particularly important in that the absolute displacement rates in CTR materials are usually less than in fission reactors (the geometry of the source and first structural wall also plays a big role here).[4] Table 1 lists some typical values normalized to 1 MW/m^2.

The next important feature of DT neutron damage is the primary knock-on atom (PKA) energy. Figure 3 illustrates that the number of PKA's with energy greater than some energy E_n is almost always greater for a fusion neutron spectrum than for a fission spectrum. The difference is particularly pronounced above 500 keV. There is evidence that certain processes in metals are influenced by the energy of the PKA's. For example,resolution of precipitates in Ni-Al alloys is quite pronounced after 2.8 MeV nickel irradiation[5] whereas the particles actually grow after electron irradiation[6] (the PKA energy is $\lesssim$ 50 eV in the latter case). The production of vacancy loops inside of displacement cascades is also increased as the PKA energy is increased.[7] Therefore, it is important to know not only the rate at which atoms will be displaced in a fusion neutron environment, but also how they are displaced with respect to the actual region of the displacement spike. This will be important later when we discuss simulation techniques .

The next most important primary reaction is that producing the insoluble gas, helium. Figure 4 shows how the helium cross section varies with neutron energy and selected values for CTR materials are quoted in Table 1. We have also plotted in Figure 4 the typical fusion and fission neutron energy spectra. Note that while the fusion spectra extends well into the (n,α) reaction energies, the fission spectra has only a very few neutrons in that energy regime. Consequently, the helium production rates in fusion reactors can be extremely large and helium is known to be particularly devastating to mechanical properties in metals at very high temperatures.[8]

Transmutations which lead to solid elements are very system and spectrum dependent and we will discuss that later in our discussion of secondary responses.

2.2 Secondary Responses

The direct transfer of energy from the incident particles to the surface atoms or

TYPICAL DAMAGE ENERGY FUNCTIONS
FOR POTENTIAL FUSION REACTOR MATERIALS

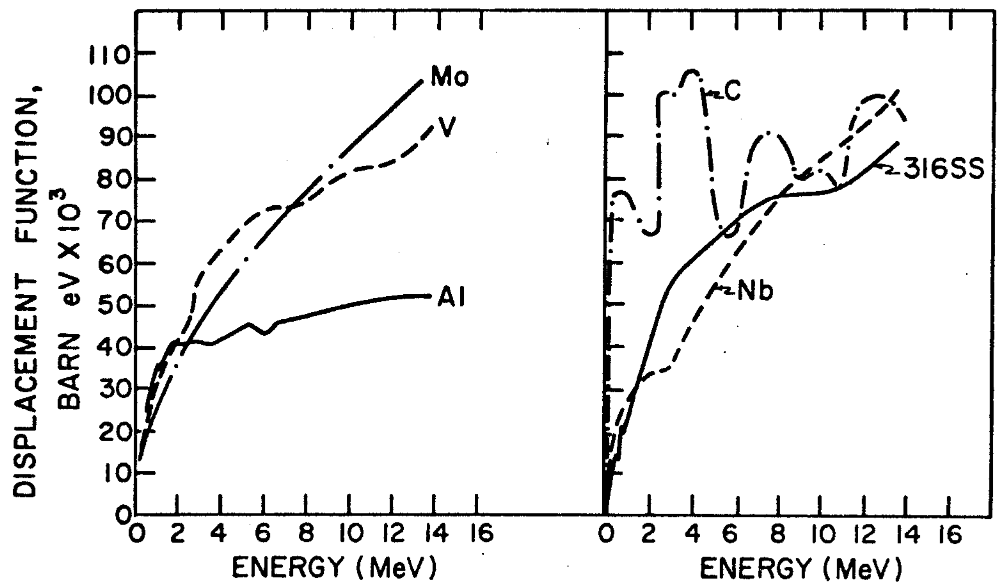


FIGURE 2

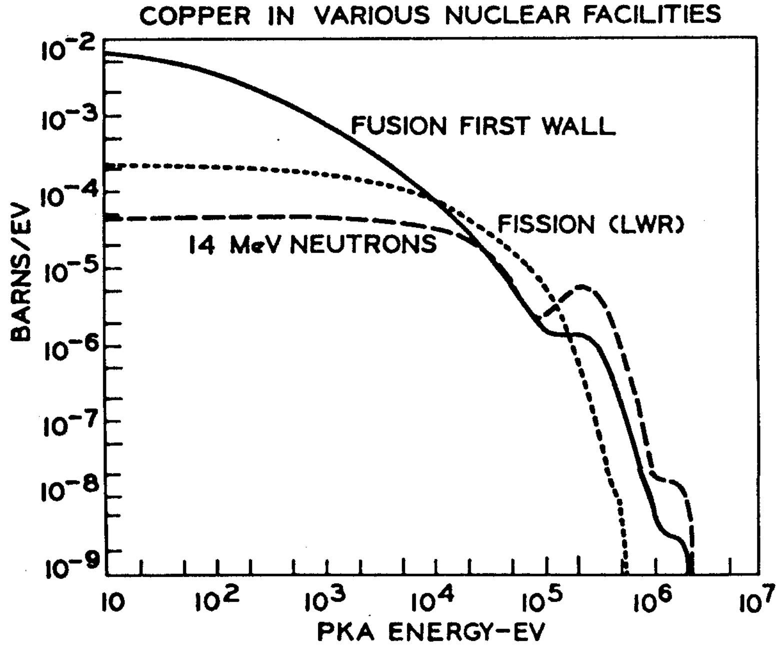


FIGURE 3

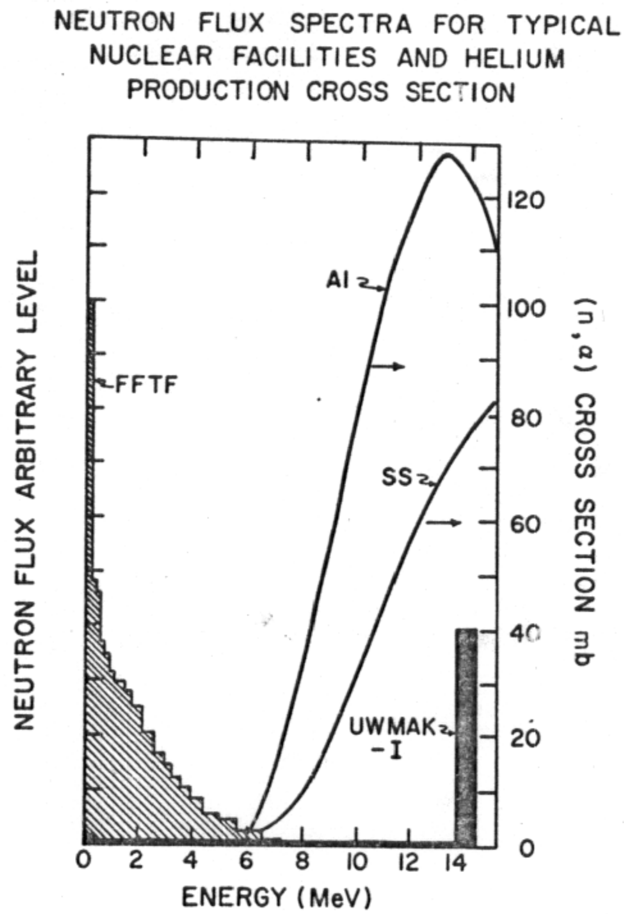


FIGURE 4

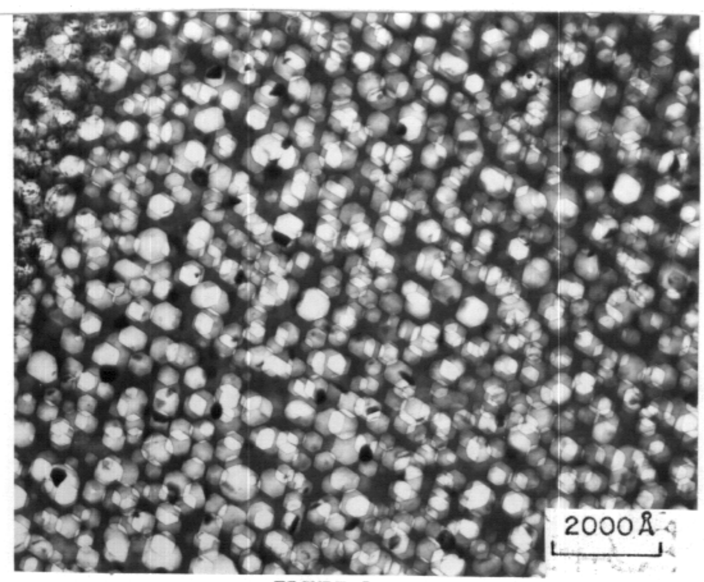


FIGURE 5

Solution treated 316 SS (fuel pin cladding) irradiated at ~525°C to 1.3 x 10^{23} n/cm^2, swelling ~25%. Courtesy H. R. Brager, HEDL the indirect transfer of energy via focussing collisions is known to cause atoms to be sputtered off the surface. The rate of wall erosion due to neutron sputtering can be expressed as follows:

$$\frac{\Delta d}{t} = \frac{S_n \phi}{N_A}$$

where Δd is the wall thickness reduction

t = time
S_n = neutron sputtering coefficient
ϕ = flux of neutrons
N_A = atomic density

For $N_A \sim 8 \times 10^{22}$ atom/cm^3
$\phi = 2 \times 10^{14}$ n/cm^2/sec per MW/m^2.

$$\frac{\Delta d}{t} = 0.16\ S_n \text{ cm per MW-yr/m}^2 \ *$$

Recent measurements of S_n vary from 10^{-3} to 10^{-5} for 14 MeV neutrons[9] and even at the highest rate, the rate of wall thinning is 2 microns per MW-yr/m^2. We shall see later that the maximum wall loading anticipated for a 10 year EPR operation time is ~10MW-yr/m^2 and hence less than 0.02 mm will be removed during the component's lifetime.

The production of voids in metals as a result of irradiation has been studied for almost 10 years. There are several good references which summarize the progress in the field and the reader is urged to consult them.[10-11] At first glance, the production of equal amounts of vacancies and interstitials should not result in any volume change. However the high mobility and extreme insolubility of interstitials combine to make the formation of interstitial loops energetically favorable. These loops in turn attract more interstitials than vacancies so that in a very short time there is an excess of vacancies in the matrix. Given sufficient nucleation sites and high enough temperature for the vacancies to move, the vacancies precipitate into three dimensional aggregates (voids) which act as further sinks for more vacancies. See Figure 5 for an example of voids in stainless steel[12] Swelling values of over 100% have been observed thus far and almost every known structural alloy shows some measure of swelling.

The important parameters in swelling are listed below.

- Temperature - It appears that the irradiation temperature must be above 20% and less than 50% of the absolute melting point (T_{mp}). Below 40% of T_{mp} the growth of voids is limiting and above 40% the nucleation is limiting.
- Stress - There appears to be little stress effect at less than 40%T_{mp}, but above that value, tensile stress lowers the critical radius required for nucleation and enhances swelling.
- Helium Gas - Some evidence suggests that it has little effect on the total swelling at low temperature but it can significantly alter the number density of voids. Above 40% T_{mp}, He gas stabilizes void nucleii and greatly enhances swelling.

* using both sides of the first wall

- Alloy (Transmutations) - In general, the more pure the metal, the higher the swelling under given temperature and damage conditions; therefore, generation of impurity atoms can reduce swelling.
- Total Damage - The swelling of most metals is proportional to the number of displaced atoms to the n^{th} power. In some metals there is an incubation period where critical void nucleii size must be attained and this may be as high as 1-10 dpa in alloys or as low as 0.01 dpa in pure metals. After the incubation dose is exceeded most metals swell with n = 0.8 to 1.2 (See Figure 6 for typical behavior in austenitic steels).[13]

The significance of all of this to near term Tokamak fusion reactors is that as we progress into the electrical production mode,the operating temperature of most structural alloys will have to be raised to above 0.25 T_{mp} and well into the void formation regime. The use of high pressure coolants and rather large thermal stresses will tend to accelerate swelling above $0.4T_{mp}$. As the wall loadings are raised, significant helium gas generation will occur and enhance high temperature swelling. As the solid transmutation products build up it is possible that swelling will be reduced. Since the useful component lifetime will be inversely propertional to the swelling rate it is imperative that a more fundamental understanding of this phenomena be developed to allow long term safe operation.

Dimensional instabilities can also be caused by the formation of helium filled gas bubbles which grow in the presence of an excess of vacancies by balancing the internal gas pressure, P, with the surface tension as follows:

$$P = \frac{2\gamma}{r}$$

where γ = surface energy
r = radius of the bubble.

The swelling induced in a solid under this equilibrium situation is simply:

$$\frac{\Delta V}{V_o} = C_{He}\left[\frac{rkT}{2\gamma} + b\right]$$

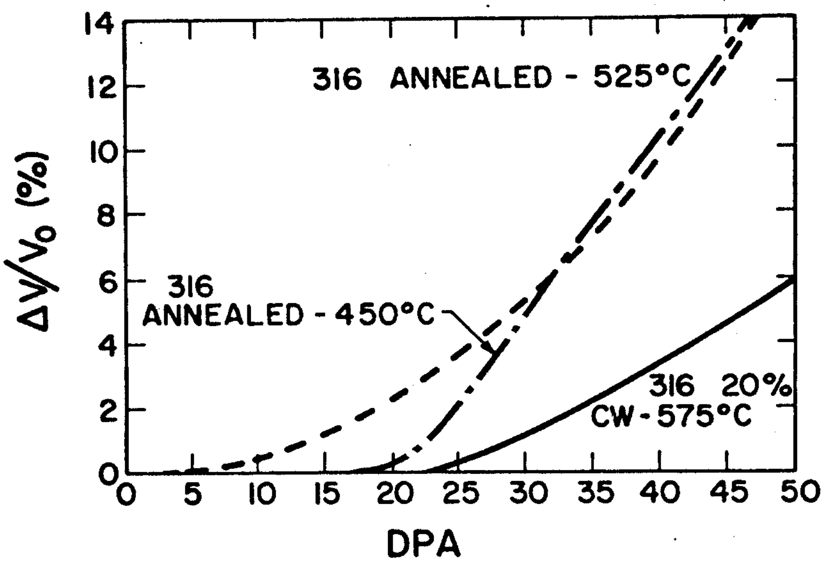
where C_{He} is the number of helium atoms per unit volume
k is the Boltzmann's constant
T is the temperature
b is the Van der Waal's constant

For typical values of the materials constants (γ = 1000 ergs cm^{-2}, b = 4 x 10^{-23} cm^3 atom $^{-1}$) the above expression for Be transforms to

$$\frac{\Delta V}{V_o}(\%) = C_{He}[6950\ rT + 4]\ (1.236 \times 10^{-4})$$

where r is expressed in Å, T in °K, and C_{He} is in appm.

Figure 7 shows how the theoretical swelling depends on various levels of helium in Be. At small bubble radius the swelling is low because of the high surface tension. As



TYPICAL SWELLING IN AUSTENITIC STEELS

FIGURE 6

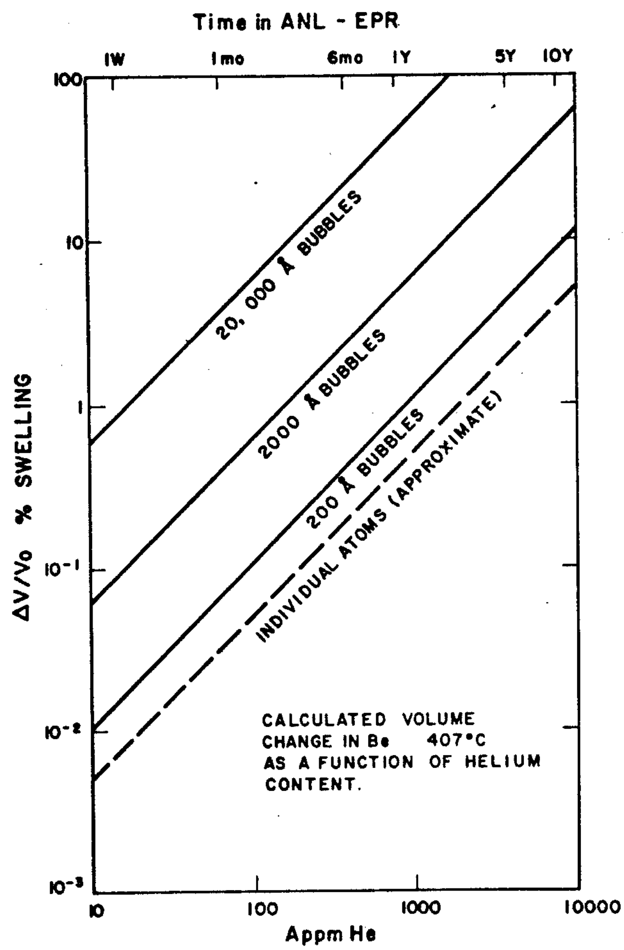


FIGURE 7

the bubble size increases the number of vacancies per gas atom increases, and the swelling is enhanced. Temperature has little intrinsic effect on the swelling but the bubble mobility is enhanced at high temperature promoting coalescence and hence large bubble radius. The significance of this particular example is that Be has been proposed [14] as a coating on the first wall to protect the plasma from contamination and if it undergoes large dimensional changes, it may not adhere to the metallic substrate.

Some non metals like graphite undergo another type of dimensional instability directly connected to the anisotropic crystal structure [15-17] Vacancies and interstitials tend to precipitate on different crystallographic planes which makes the single crystals grow anisotropically and even randomly oriented, fine grain material will swell. The general behavior is for the polycrystalline material to shrink initially while all of the as fabricated porosity is being filled. Eventually this process saturates and the specimens begin to grow very rapidly in the so called runaway mode (Figure 8). The damage level to achieve this is 1-10 dpa and the cross over point ($\Delta \ell=0$) appears to decrease with temperatures below 1000 °C and increase with temperatures above 1000 °C.

The significance of this phenomena for tokamak reactors is that there have been several proposals to reduce the impurity problem by lining the surface facing the plasma with carbon either in a sheet or coating form. If the carbon grows during irradiation it could buckle, tear, or crack the layers, or cause the coatings to peal off thus eliminating the effectiveness of the protection. One way to alleviate this problem is to use the carbon in a loose 2 dimensional weave which could retain considerable flexibility under large dimensional changes. [18]

Resistance changes in metals irradiated at low temperature has been studied for more than 20 years. It is a well established fact that, at low fluences, the amount of resistivity increase is directly proportional to the number of defects produced, inversely proportional to the temperature of irradiation, and moderately dependent on the type of particle used to damage the specimens. The numbers range from 2-15 micro-ohm-cm per % Frankel pairs at low temperature until relatively high damage levels ($\sim 10^{-4}$ - 10^{-3} dpa). [19] Saturation at damage levels higher than this is due to overlapping damage zones and annealing of existing damage by the displacement spikes (See Figure 9).

The practical significance of this effect in tokamak fusion reactors is that neutrons leaking out of the blanket and shield zone could increase the resistance of normal magnets or increase the resistance of the stabilizer in superconducting magnets. The effect is not so important at room temperature and above, but can be quite serious at 50° K and lower. The higher resistance increases the I^2R losses to the cryogenic cooling system and could eventually require that the magnets be annealed (at RT to 300 ° C) to remove the damage.

Irradiation of superconducting filaments is known to reduce T_c and J_c in some alloys. Nb_3Sn is particularly sensitive (Figure 10) in the 10^{-3} to 10^{-2} dpa range whereas the NbTi system is relatively immune up to about $\sim 10^{-2}$ dpa. The practical significance here is that the continued use of S/C coils in a radiation field could cause the coil to

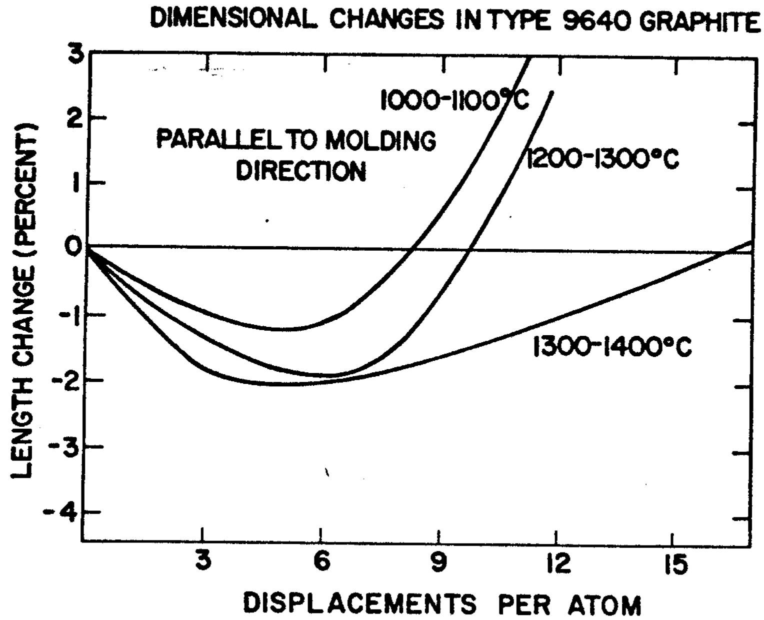


FIGURE 8

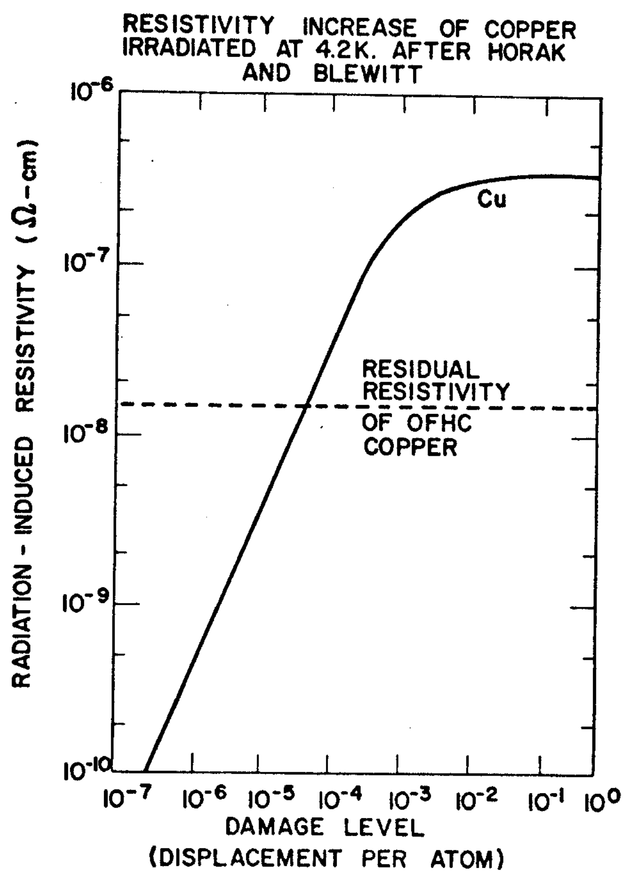
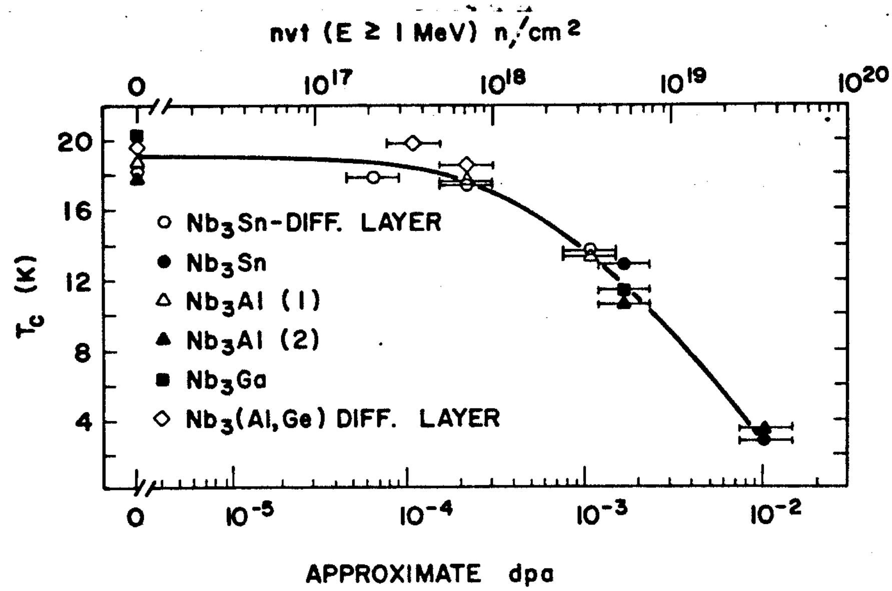


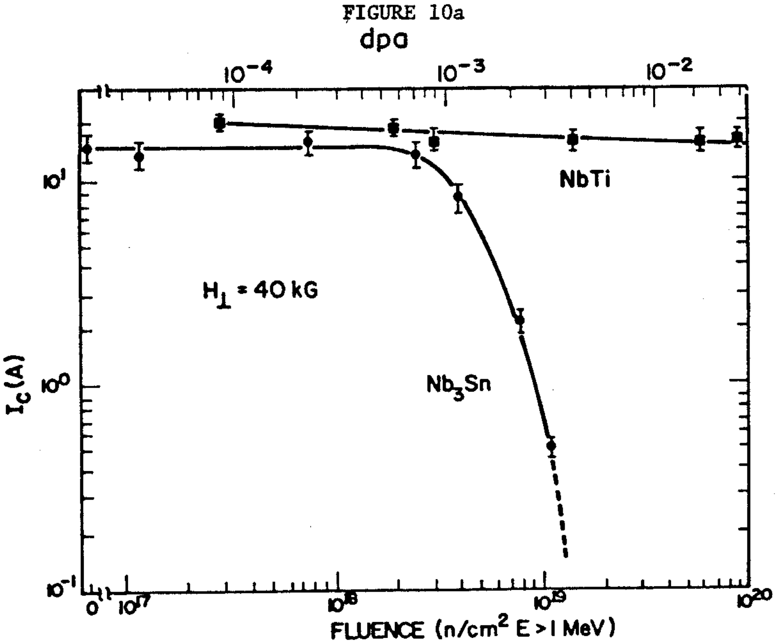
FIGURE 9



EFFECT OF NEUTRON IRRADIATION ON CRITICAL TEMPERATURE OF A-15 SUPERCONDUCTORS

(AFTER SWEEDLER et al)

FIGURE 10a



EFFECT OF NEUTRON IRRADIATION ON THE CRITICAL CURRENT IN NbTi AND Nb_3Sn.

FIGURE 10b

go normal due to drops in T_c or J_c, thus jeopardizing the integrity of the magnet and the continued operation of the reactor.

It is well known that irradiation produced defects in metals can act as significant barriers to dislocation motion, thus causing the metal to become stronger. These clusters can range all the way from small loops at low temperatures to voids at high temperatures. The general effect is to raise the yield strength to as much as 3-5 times the unirradiated value. While such strengthening appears beneficial on the surface, it is usually accompanied by a drop in ductility which is detrimental. (See Figure 11)

The hardening of the matrix in the grains of metals causes the deformation under stress to take place along the grain boundaries. When the deformation is localized in this manner, premature failure can occur. The generation of large amounts of helium, which has a tendency to collect on the grain boundaries, tends to aggravate an already serious situation. The bubbles interfere with grain boundary sliding causing micro cracks to form on the grain boundaries, which eventually link up to cause intergranular fracture. This fracture can occur at uniform elongations of <1% (see Figure 11, 12) and a reactor which is made of metal in this state is very subject to crack formation. Any rapid loading or unloading of the stresses could cause brittle fractures to occur, destroying the vacuum conditions and jeopardizing the integrity of the blanket structure.

The significance for fusion reactors is that because of the high helium generation rates, (Figure 4 and Table 1) the probability of brittle fracture at high temperature is very large. This puts an added penalty on high temperature ($>0.4\ T_{mp}$) operation for DT fusion devices and also has a great bearing on the lifetime of a first wall component.

The plastic deformation of materials at high temperatures, under high stresses below the elastic limit, is well known and called thermally induced creep. This phenomena can also occur during neutron irradiation because of the large concentration of vacancies and interstitials available for dislocation motion. The thermal creep rate is extremely temperature dependent but the irradiation induced creep is relatively independent of temperature (see Figure 13). The consequence of this is that above $0.5\ T_{mp}$ thermal creep is dominant and below that temperature irradiation creep is dominant. The level or irradiation creep is proportional to the displacement rate and can be beneficial or detrimental for fusion reactors. On the one hand, it will tend to relieve stress concentrations built up because of differential swelling, thermal expansion or fabrication difficulties. On the other hand, the overall dimensions of the systems will change making insertion or extraction of some components very difficult. Warping of certain coolant channels could also adversely effect the temperature profiles in the blankets. Fig. 13 also shows how the irradiation creep rate depends on total damage.

The constant cycling of temperature or stress levels can cause the premature failure of certain metallic components by a process called "fatigue." The generation and propogation of cracks at alternating stress and stain levels in the elastic regime is well known and the number of cycles to cause failure in metals usually decreases as the magnitude of stress or strain is increased. Figure 14 shows how the fatigue failure in 316 SS at 593 °C is affected by the strain range per cycle. [14] The effect of irradiation on fatigue life is rather unclear at this time but in some cases, the generation of dislocation loops tends to reduce crack propagation and extend useful lifetime.

The significance of this phenomena is readily apparent when we remember that tokamak devices normally operate in a pulsed mode,with as many as 100,000 or more cycles per year anticipated for some EPR designs.

The transmutation reactions which occur in practically all neutron irradiated elements will convert some stable isotopes into unstable isotopes. The decay of these isotopes can produce large radiation fields which prohibit unprotected personnel from doing maintenance on the reactor components. The total induced activity levels range from 1-5 curies per thermal watt of energy generated and can take years to decay (Figure 15). Thus, extremely conservative design policies need to be followed to insure long term safe operation and these policies may tend to penalize one material much more than another (i.e. steel versus vanadium).

The decay energy of the radioactive species will generate heat, which, if not properly accounted for in fission reactors,can actually cause some metallic components to melt. However, the afterheat density in fusion reactors of ~1 watt per cm^3, is usually at least an order of magnitude lower than for fission reactors and normally is not a serious problem.

Finally, the production of some isotopes in certain metals can cause significant changes in their composition and formation of secondary phases. For example, the solubility of Zr in Nb is ~10% at normal operating temperature and in a typical CTR neutron spectrum the production rate is 0.1 to 0.2 atomic percent per $MW\text{-}yr/m^2$. The compositional changes may be particularly important if the recently observed irradiation induced precipitation phenomena are a general rule of CTR alloys [20-21]

3. Summary of Selected Operating Parameters and Damage Conditions for Near Term Fusion Reactors

As noted in the introduction, we are assessing the next three generations of reactors while they are in a very fluid state of design and any parameters which are quoted must be carefully referenced. Future readers of this paper should be sure that key parameters have not changed because of a reassessment of the very problems we are addressing. It is not the purpose of this paper to completly review the future tokamak reactors, but rather to highlight the potential materials problems associated with neutron damage. Therefore, we have included in Table 2 an abbreviated list of pertinent parameters for the analysis to be performed in Section 4. We will only consider devices that will burn a substantial amount of D&T (e. g., not JT - 60).

The dates of first DT operation in tokamaks vary from 1983-6 for the first generation of devices to 1991 for the EPR's and 1998 for the DPR (logic III, USERDA [22]). Thermal power levels during the burn will be quite low in TFTR (20 x 10 MW) and increase to 400-600 MW for the EPRs and probably ~1500-1700 MW for the DPRs. Taking into account the fraction of the energy produced by neutrons and wall area,we find that the first wall neutron loadings vary from a maximum of 0.25 MW/m^2 for the first generation devices to 0.6-0.8 MW/m^2 for EPRs and probably 1-3 MW/m^2 in the DPRs. When the burn cycle and plant factors are included we find the integrated first wall loadings range from a low of $10^{-5} MW\text{-}y/m^2$

EFFECT OF NEUTRON IRRADIATION ON THE YIELD STRENGTH AND DUCTILITY OF 304 STAINLESS STEEL - (AFTER FISH ET AL. - HEDL)

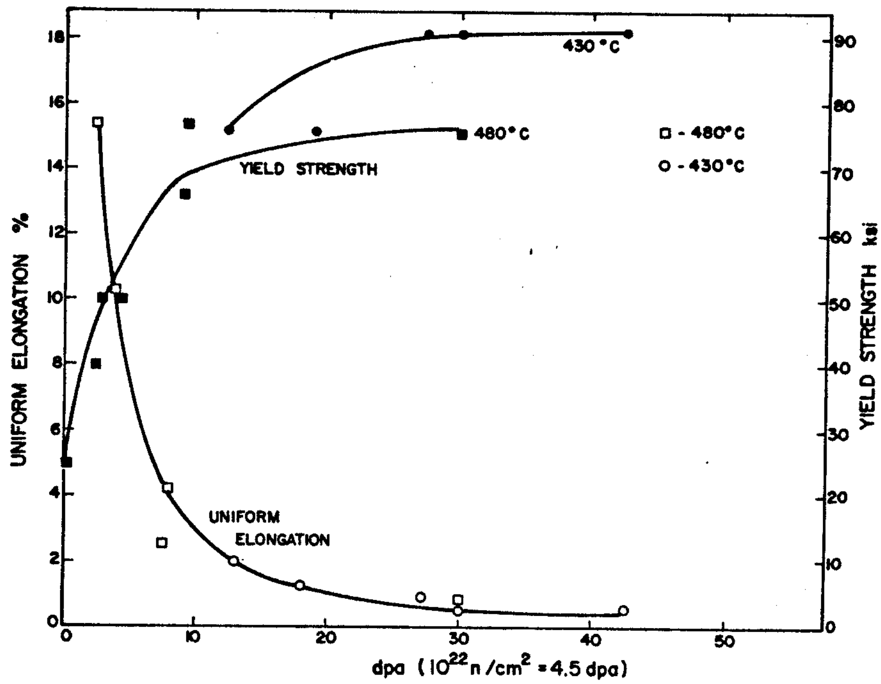


FIGURE 11

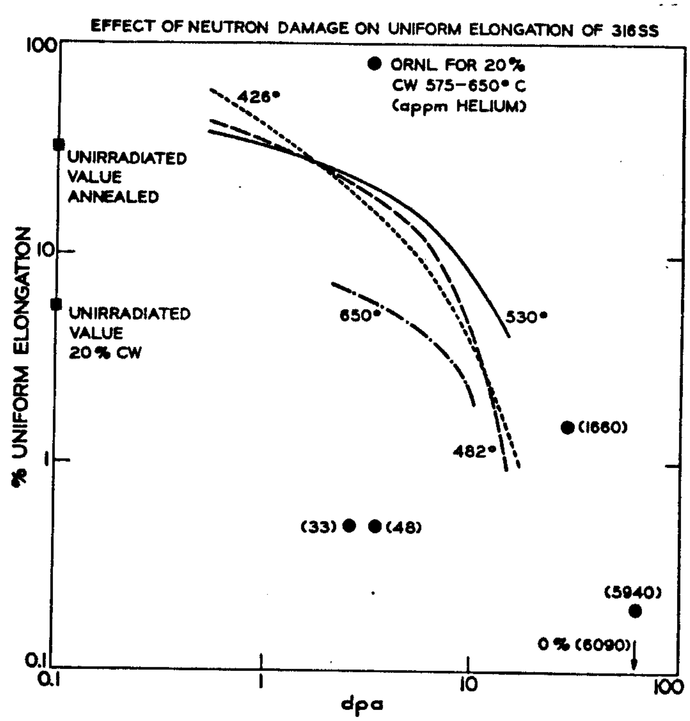


FIGURE 12

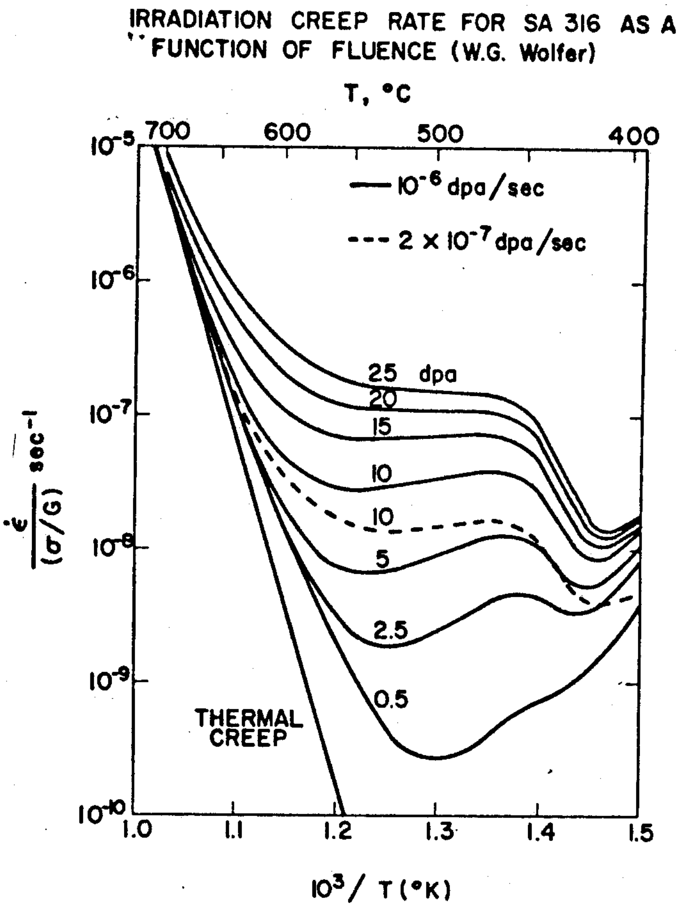
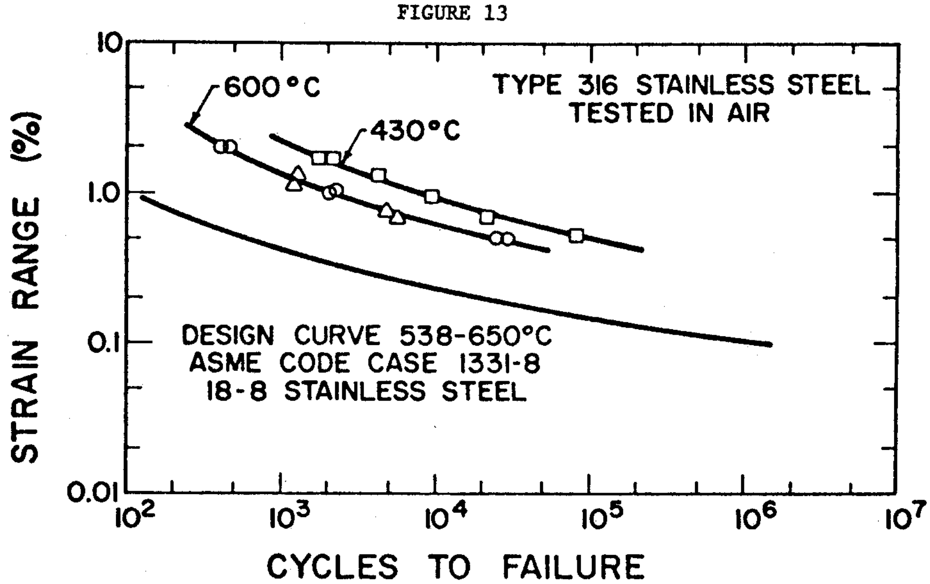


FIGURE 13



FATIGUE LIFE OF ANNEALED 316 STAINLESS STEEL AT 593°C UNDER VARIOUS TEST CONDITIONS ($\dot{\epsilon}=4 \times 10^{-3} s^{-1}$).[14]

FIGURE 14

FIGURE 15

Radioactivity of CTR Blankets After Shutdown

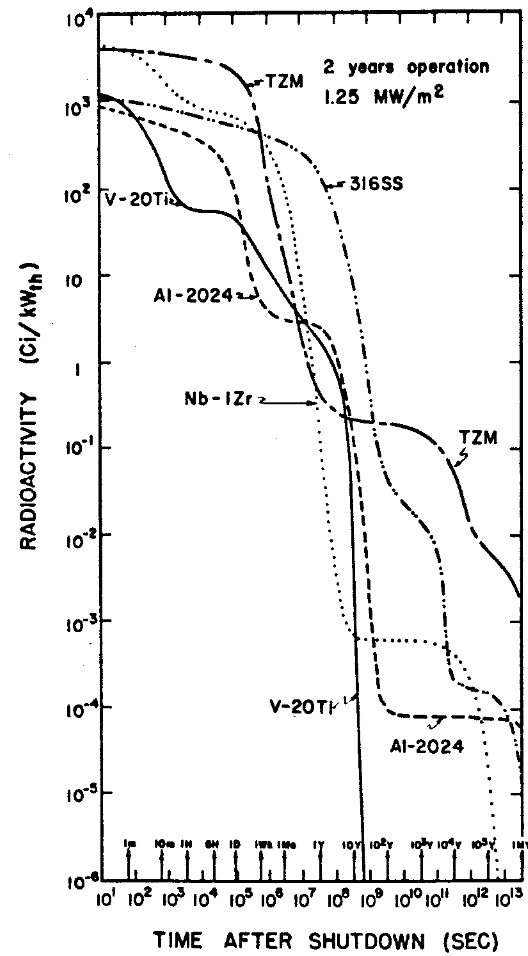


FIGURE 16

MAXIMUM PROJECTED DISPLACEMENT DAMAGE IN METAL STRUCTURES OF NEAR TERM TOKAMAK FUSION REACTORS.

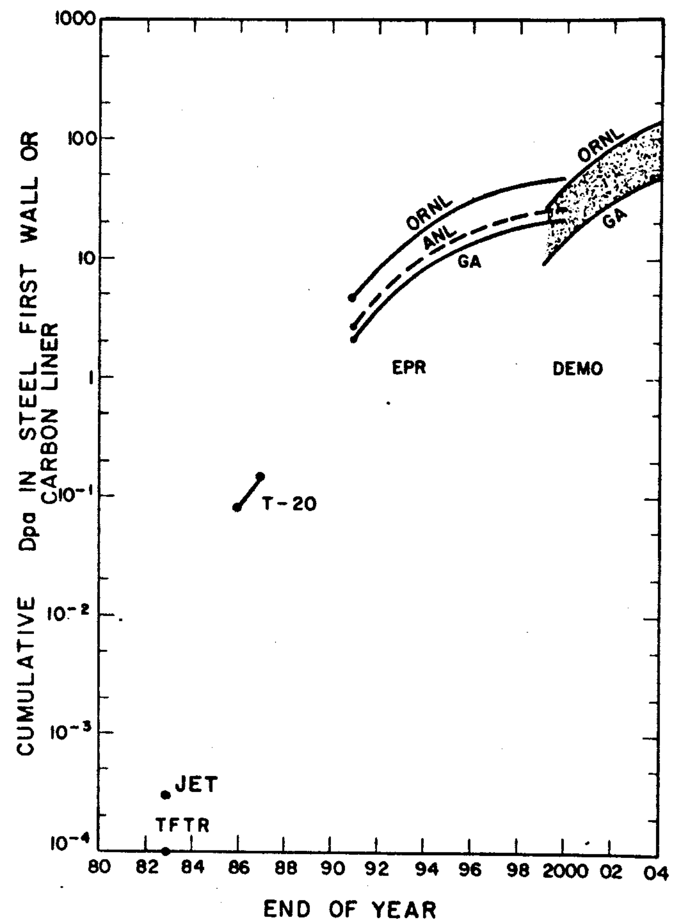


Table 2

Summary of Selected Parameters of Near Term DT Tokamak Systems

Parameter	First Generation (23)			Experimental Power Reactors			Demonstration Power Reactors	
	TFTR	JET	T-20	ANL	GA	ORNL	GA	ORNL
Reference				(24)	(25)	(26)	(27)	(28)
Est. Date DT OP	1983	1983	1986	1991	1991	1991	1999	1999
Inst. Power-MW_t	10	15	55-140	638	410	410	1660	1700
First Wall Area-m^2	110	220	400	592	~390	600		350
Neutron MW/m^2	0.1	0.05	0.25	0.56	0.83	0.55(e)	1.2	3.0
Burn Time-sec	1	20	5-20	55	105	100	517	1200
Downtime-sec	300	600	240	15	20	15	60	50
Max Plant Factor	1-04	6-04	0.03	0.5	0.2	0.7	0.7	~0.7
Cycles per Year	1000	5000	5+04	2.3+05	5+04	1.9+05	3.8+04	1.8+04
14 Mev n $cm^{-2}y^{-1}$	2+16	4+16	1+19	4+20	2+20	1+21	$1x10^{21}$	$3x10^{21}$
Total n $cm^{-2}y^{-1}$	NS	NS	4+20	6+21	4+21	1+22	1.2+22	NS
Est. DT Reactor Life-y	1	1	2	10	10	10	25	25
First Wall Exp. MW-y/m^2	1-05	3-05	0.015	2.5	1.7	3.9	21	53
Low Z Liner	None	None	None	Be	SiC/C	C	SiC	NS
Liner T_{max} °C	--	---	---	380	1450/1650	1780	1570	NS
Met. Struc.	305SS	Inconel	SS	316SS	316SS	316SS	Inconel	Adv. SS
Coolant				H_2O	He	H_2O/He	He	Metal Salt
Coolant, Inlet/Outlet °C				40/310	325/575	40/121(H_2O) 200/370(He)	350/650	NS
T_{max} First Wall-°C	80	50	100/500	500	600	125/540	700	400
Stress 1st Wall-MPa	---	---	---	69	200	NS	200-400	NS
Blanket Materials				SS/H_2O	SS/C	SS/C/K	(a)	SS/K/Na/Li

Table 2 (con't.)

	First Generation			Experimental Power Reactors			Demonstration Power Reactors	
	TFTR	JET	T-20	ANL	GA	ORNL	GA	ORNL
Blanket Thickness-cm				28	25	52	70	NS
Shield Materials				$SS/B_4C/Pb$	$SS/B_4C/LiH/Pb$	$SS/Pb/H_2O$	(b)	NS
Shield Thickness				58/97	40/100	48/48	70/70	NS
Inside/Outside, cm							NS	NbTi or Cu
Magnet Materials-TF	Cu	Cu	Cu	NbTi/Cu	NbTi/Cu	$NbTi/Nb_3Sn/Cu$		
Magnet Materials-VF	Cu	Cu	Cu	"	Cu	Cu/NbTi(d)	Cu(c)	NbTi or Cu
Magnet Materials-OH	Cu	Cu	Cu	"	NbTi/Cu	NbTi/Cu	NS	NS
T_{max} Magnets-^oK-TF	RT	RT	RT	3.0	4.2	4.0	NS	4.2 or RT
T_{max} Magnets-^oK-VF	RT	RT	RT	4.2	313(c)	RT/4.2(d)	NS	4.2 or RT
T_{max} Magnets-^oK-OH	RT	RT	RT	4.2	4.2	4.2	NS	4.2 or RT
DPA rate-^{1}Liner-y^{-1}	--	--	--	NS	---	NS	--	--
DPA rate 1^{st} Wall-y^{-1}	1-04	3-04	0.07	2.8	2.2	4	9	23
DPA rate TF coil-y^{-1}	--	--	--	8-06	3-05	2-04	5-05	NS
appm-He-y^{-1}-liner	--	--	--	780	210	1200	1600	--
appm-He-y^{-1}-1st Wall	0.002	0.006	1.4	54	37	129	110	420
MRad-y^{-1} -Insul.	--	--	--	40	100	100	---	---

$1 + .04 = 1 \times 10^{+4}$

NS - not stated

NA - not applicable

a) $Inconel/C/Li_7Pb_2/Li_4SiO_4$

b) $ss/B_4C/C/Pb/Al$

c) F coil

d) Cu shield VF coil, NbTi Trim VF coil.

e) Based on personnal communication, J. Clarke per year for first generation DT devices to 0.2 to 0.4 $MW\text{-}yr/m^2$ per year for EPRs and ~1-2 $MW\text{-}yr/m^2$ per year for DPRs.

The estimated lifetime of the reactors (using DT) are somewhat arbitrary, but we have assumed a 1-2 year lifetime for the first generation devices with tritium, 10 years for EPRs, and 25 years for DPRs. These assumptions yield total first wall exposures of 10^{-5} $MW\text{-}yr/m^2$ for first generation devices, 2-4 $MW\text{-}yr/m^2$ for EPRS and 20-50 $MW\text{-}yr/m^2$ for DPRs.

Most EPR designs now call for low Z liners to reduce plasma contamination. These range from Be in the ANL reactor to SiC and C in the GA design and carbon in the ORNL design. Specific proposals for low Z liners have not been made for the DPRS, but they will probably be even <u>more</u> necessary there. These liner temperatures are envisioned to range from 400°C for Be to ~1500°C for SiC and 1700-1800°C for carbon.

The metallic structure for the first generation reactors includes various austenitic alloys such as 305 SS and some specific Inconel compositions. The EPRs uniformly employ 316 SS and the DPRs are likely to utilize some advanced Fe-Cr-Ni alloy (i.e., Modified 316 SS, Inconel 718). A common coolant in all the first generation and EPR reactor designs is water and helium coolant is also used in the GA and ORNL EPR design. The DPR is likely to use liquid metals or He. The maximum first wall structural temperature is <100°C in the first generation designs (except for some parts of T-20) and ranges from 380 to 650°C in the EPR design. The ORNL design cools the first solid wall to 125°C and parts of the blanket immediately behind that operate at high temperatures (up to 540°C). The DPR reactor designs run the first wall at 400-700°C. Stresses in first walls were not stated in all designs, but in the higher temperature EPRs they are 70-200 MPa.

There are other materials besides the structure and coolant in the blanket. The EPR designs use C, K, and borated water to reduce the neutron energy. The blanket thickness ranges from 20 to 52 cm in all the designs.

The shields of the near term reactors contain Pb, B_4C, steel, C, and some borated water. The shield thicknesses in these designs range from 50-100 cm thick.

All the magnets for the first generation designs are copper and it is only when one gets to the EPR and DPR designs that superconducting TF, VF, and OH coils are used. The GA EPR design still uses a normal F coil. The superconductors are NbTi with varying amounts of copper stabilizer and the ORNL design even calls for some Nb_3Sn. Operating temperatures of the normal coils are ~ 40°C while the S/C coils run at liquid helium temperatures.

The maximum damage rates in the liners of the EPRs are on the order of ~2-4 dpa/yr for C and SiC (no values for Be have been calculated). The maximum helium production rates are ~1200 appm per year for carbon and ~200 for SiC and ~800 for Be. The values would be somewhat higher for the DPRs.

The first wall (assuming steel) of the first generation reactors will suffer only modest damage levels up to a maximum of 0.07 dpa per year while those in the EPRs experience from 2-4 dpa per year with an associated helium production of 40-80 appm/yr.

These values are increased to 10-20 dpa per year in the DPR in addition to 100-400 appm He per year in the first wall.

Damage to the TF coils is very low in the first generation designs ($<<10^{-7}$dpa/year) and the TF coils of the EPR and DPR designs may experience 10^{-5} to 10^{-4} dpa/year. Calculation of damage to the organic electrical insulators has been performed and it is found to range from 40 to 100 MRad/year.

The time dependence of the displacement and gas production damage in the low Z liners and steel first walls is shown in Figures 16 and 17 where cumulative values are displayed. It can be seen that significant displacement damage in the metallic first walls does not occur until 1991. The damage in the ORNL EPR is about twice as high as the ANL and GA design. For reference, note that it takes ~1 year to generate ~20 dpa in the current fast neutron test reactors which is essentially end of life for the ANL and GA designs and equal to <5 years in the ORNL design. The damage level in the ORNL DPR exceeds that in the ORNL EPR in approximately 2 years after startup and it will reach the end of life level for the GA-EPR design in approximately 1 year.

The major point about low Z liners is that the Be coating in the ANL design could contain ~800 appm He after one year and ~8000 appm after 10 years. The helium in the C liner of the ORNL design could approach 12,000 appm after 10 years of operation and the SiC of the GA design may contain 2000 appm after 10 years.

The displacement damage in the TF coils will reach ~8×10^{-5} dpa in the ANL design and 2×10^{-3} dpa for the ORNL design after 10 years of operation. The damage rate in the normal F coils of the GA design is approximately 0.01 dpa/year and would amount to 0.1 dpa at the end of life.

To put this data into perspective, we have plotted in Figures 18, 19 and 20 the current status of data for these materials. It is noted in Figure 18 that both of the EPR designs that use carbon as a liner material are using it in a damage-temperature regime that has never before been explored. The GA carbon liners, if left in for the life of the reactor, would achieve an exposure about as high as the highest reported in the literature and at temperatures of ~200-300°C above the highest reported data. The carbon in the ORNL design would be subjected to about the same as the highest reported exposure to carbon up to now at temperatures of ~300°C above the highest tests to date.

The situation for the displacement damage in 316 SS is much better (Figure 19). Steel has been tested at high temperture to damage levels as high as required for all EPR designs. Low temperature damage data is not available.
More serious is the lack of a large body of data on high helium contents. At low temperatures, the thermal neutron production of He from Ni-59 has generated ~500 appm in some LWR cladding. As one goes to higher temperatures where fast reactors operate, the amount of helium generated drops considerably. If this was the only data available, the EPR designs would exceed it by factors of 10-100. Fortunately, limited high temperature data is now available in thermal reactors (shown in Figure 19). This information should help to determine the effect of helium on dimensional stability and mechanical integrity. Unfortunately, the helium tends to enhance swelling (typical

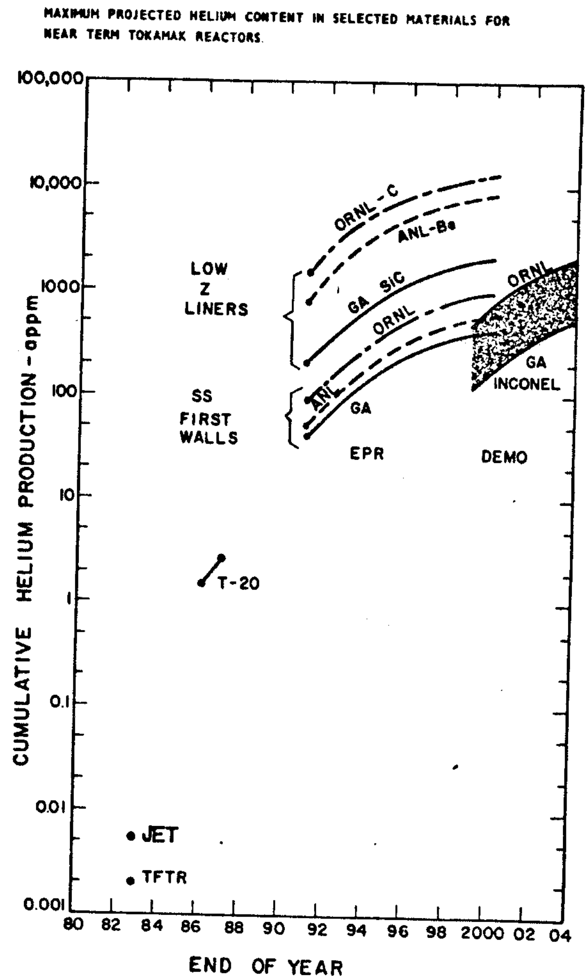


FIGURE 17

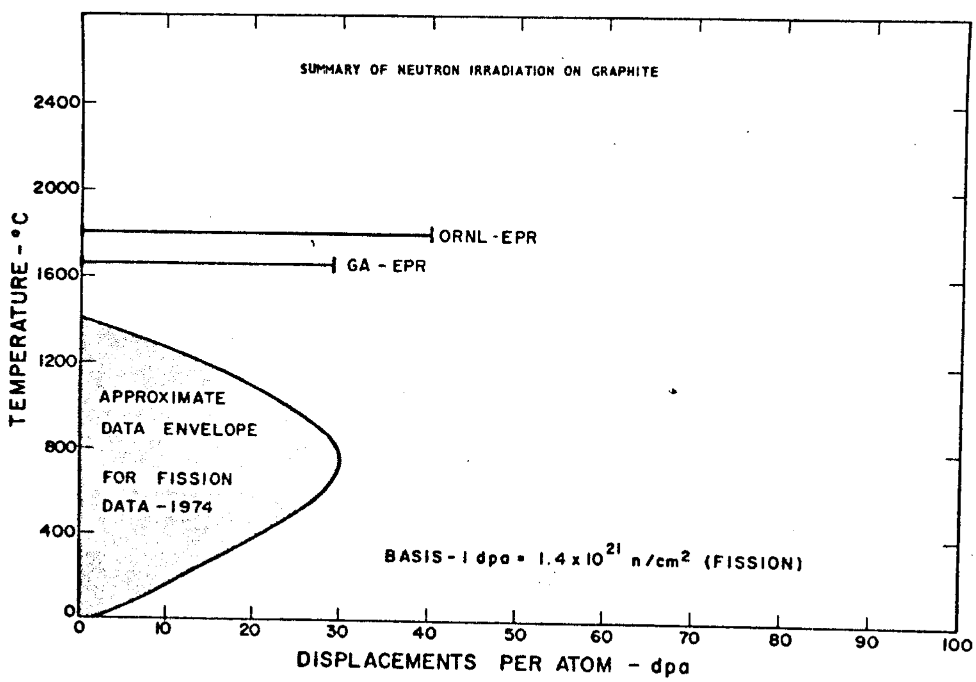


FIGURE 18

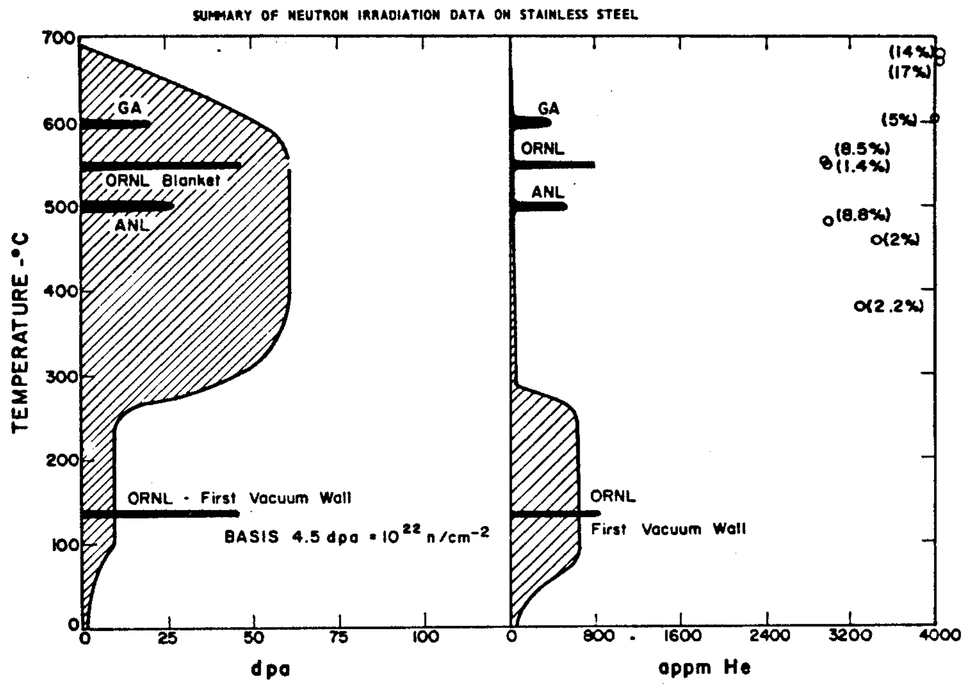


FIGURE 19

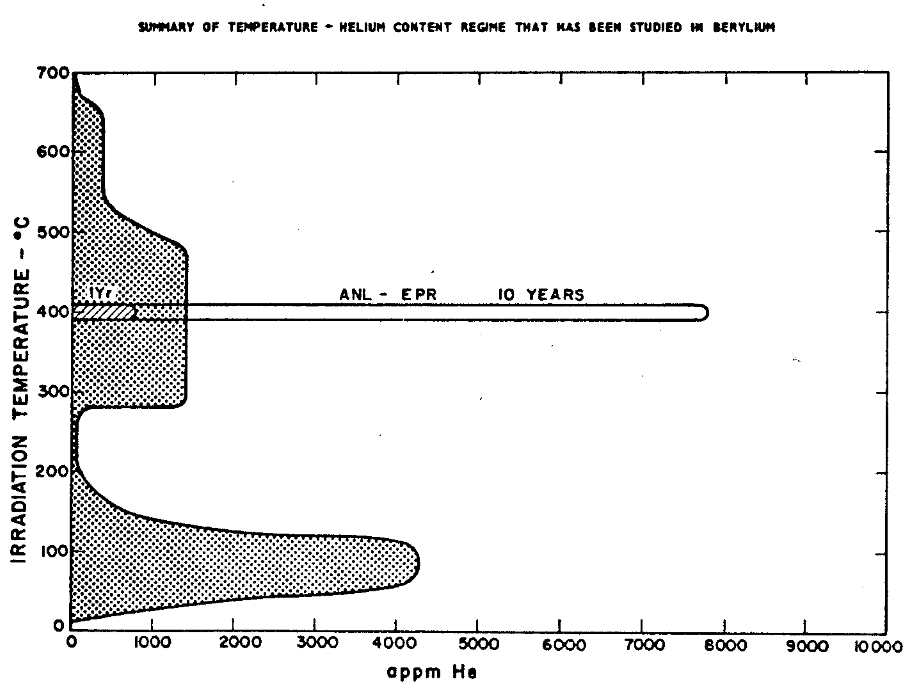


FIGURE 20

values are shown next to high helium data points in Figure 19) and reduce the ductility as we have seen.

Finally, Be metal has been irradiated to ~4000 appm (~5 years in ANL EPR) at temperatures ranging from 40-100°C. There is very little data from ~100 to 300°C and then information is available from 300-500°C for up to 1500 appm He. There is little information available above 700°C and limited data at 500-650°C (~400 appm He). The Be coating in the ANL-EPR design is envisioned to operate at ~400°C and will accumulate helium at the rate of 780 appm per year. Hence, the end of life will exceed 5 times the present data base.

4. Potential Problems of Neutron Irradiated Materials in Near Term Tokamak Reactors

4.1 First Generation Reactors

From the analysis in Section III it is apparent that very little, if any, materials degradation will take place in near term tokamak reactors. The low temperature, low wall loading, low plant factor, and relatively short device life combine to minimize any neutron damage effects. The only area that could present problems might be that of electrical insulators in or near the first wall. So little is known about the fundamental effects of 14 MeV neutrons (or even fission neutrons) on electrical conductivity that we hesitate to say that there will be "no" problems in the first generation tokamak reactors.

4.2 Experimental Power Reactors

The first major problem that may arise is associated with the low Z liners. In the case of Be it is the dimensional instabilities caused by the generation of large amounts of helium and with carbon it is anisotropic growth of the crystallites.

Figure 7 shows how the swelling in Be varies with the amount of helium generated and the size of the bubbles. The approximate time in the ANL-EPR is given at the top of the graph. Obviously it is important to keep the helium from agglomerating into bubbles because they then must capture many vacancies to retain equilibrium with the surface tension, $2\gamma/r$. The operating temperature of ~400°C is low enough to keep the bubble size to probably ~200 Å or lower although high stresses are known to promote bubble agglomeration. The questions to be addressed for Be are; will there be any transients, thermal gradients, or hot spots in temperature thay may cause the Be to exceed a design temperature of 400°C (and thereby promote bubble formation, movement and coalescence); what is the maximum expansion that a coating can experience and still avoid cracking, peeling or flaking; and what effect will thermal (stress) cycling have on the precipitation of Be into bubbles and promoting larger values of swelling?

The carbon "shingles" in the ORNL design have somewhat the same problems as the Be except they may go through a shrinkage stage first before progressing into runaway swelling and possible fracture. The dpa level at which this runaway swelling will occur is very much dependent on the material texture and temperature.[15-17] At 1400°C this may occur at damage levels of as low as 10-20 dpa for some graphites. Intuition would lead us to believe that this runaway swelling will be reached at a higher dpa level at 1800°C but no firm data exists and it would be safe to say there is some doubt whether the shingles will last the full lifetime of the reactor (~ 40 dpa). Further research is desperately need nere.

The pessimism for the carbon shingles in the ORNL design can be somewhat tempered in the GA design because of the lower wall loading (1.7 MW-yr/m^2 vs 4 MW-yr/m^2 in the ORNL design). This damage level of ~ 17 dpa can almost be tolerated at 1300 - 1400° C and it is anticipated that the runaway swelling at 1650° C will occur at dpa levels of > 20.

Even if the runaway swelling in carbon were no problem in the GA and ORNL designs, one might legitimately worry about the high helium generation rate (up to ~1200 appm per yr in the ORNL design) and what effect several thousand ppm Helium may have on the physical and mechanical properties of carbon. Fortunately, it appears that helium can readily diffuse out of graphite above 1200°C. Thomas, et al,[30] have shown that there is ~100% remission of implanted helium at 1200°C, Holt,et al,[31] have measured helium diffusion distances in graphite and find that randomly migrating He can travel 100 to 1000 microns in 100 sec at 1700 °C. These high diffusivities, coupled with small crystallite size (10 microns) should allow most of the neutronically generated helium to escape. Therefore, as long as the carbon is operated at high temperature (≥ 1200 °C) we would tentatively conclude that there should be no severe problem with helium build up as there is in the case of metals.

The SiC plates in the GA-EPR will suffer some linear expansion during irradiation and this may amount to 1 to 1.5% at 1350 °C.[32] Such expansions in a small plate are probably tolerable if it is unconstrained. A more serious problem may arise from the ~300 appm helium that is generated per year. If this helium does not diffuse out of the SiC (like it does in graphite at those temperatures), then there may be a considerable volume change associated with bubble formation. Such volume changes could amount to 1-10% depending on the bubble radius.

Summarizing the situation with respect to low Z coatings and liners one can say that, aside from surface effects and chemical reactions, the neutron induced dimensional changes will be the most critical problems. The effect of stress on bubble agglomeration in Be, the runaway swelling threshold for C at 1600 °C and 1800 °C, and the release of helium in SiC are all areas that must be assessed.

The next class of problems that could arise have to do with the high operating temperatures of the first metallic walls in the reactor designs. These temperatures range from 500 °C in the ANL design to 600 °C in the GA design.* The phenomena of concern here are swelling, creep, fatigue, and ductility. Fortunately, the dpa rates are relatively low in the ANL and GA EPR design (2-3 per year) such that even after 10 years the swelling values predicted from fission reactors and simulation studies amount to a few percent of less (Figure 6). However, when high helium gas generation rates are taken into account (Figure 19) the levels could be as high as 9%. There is one word of caution that needs to be inserted here. All of the previous data comes from essentially stress free material and the recent discovery of stress enhanced swelling [33] may cause even higher swelling to be produced (see Figure 21). If swelling values exceed 5%, there will probably be regions of high stress generated at joints, welds, or in regions of high swelling gradients. The situation may be even more critical in the ORNL EPR where the fluence levels are even higher.

* The very first 316SS wall to intercept particles from the plasma in the ORNL design is cooled to ~125°C. However, immediately behind that wall is a first wall running at much higher temperature (~540°C) and the neutron loading is not very much lower.

FIGURE 21 STRESS EFFECT ON SWELLING - WEINER & BOLTAX

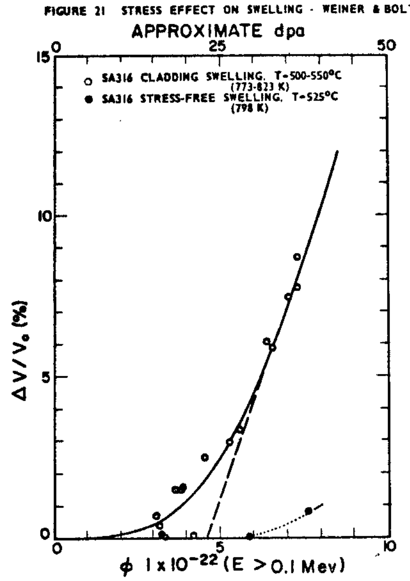
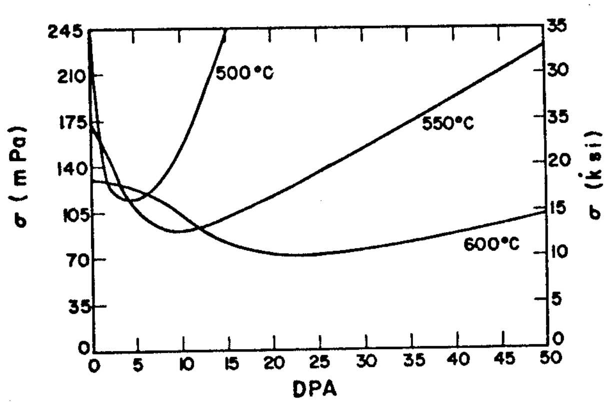


FIGURE 22 100,000 HOUR CREEP RUPTURE STRESS IN STAINLESS STEEL – Reference 34



Irradiation creep is certain to take place in the highly stressed portions of the first walls. As can be seen from Table 2, the stress levels can be 70-200 MPa. Figure 13 predicts that at 500 °C and 200 MPa, creep strains of 1-5% could occur at the end of life of some of the components (if they could stand that much strain without failing!) This will be even more aggrevated at the higher stresses levels that could occur at welds, joints or corners. These strains can act to relieve stresses built up due to differential swelling or fabrication defects, but they might also impose difficult disassembly problems on the maintenance crews. Therefore, the importance of tight dimensional stability must be carefully reviewed before the reactor begins to operate.

Another measure of the "low" stress, long term behavior of irradiated steel is its creep rupture strength for its useful life. In this case it should be ~100,000 hrs. for EPRs if the first walls are to last for the lifetime of the plant. The effect of displacement damage with only a small amount of helium generation is given in Figure 22.[34] Note that the minimum of the creep rupture stress is 110 MPa for 500 °C at ~5 dpa (1/2-1 years in an EPR), 85 MPa at 550 °C and ~10 dpa (~ 1-2 years in an EPR), or 70 MPa at 20 dpa (~ 2-4 years at 600 °C). Clearly the present EPR designs do not meet these criteria because their design stresses are 70-200 MPa. One might anticipate that the simultaneous generation of helium would aggravate this already serious situation and data is urgently needed in this area.

The fatigue problems in the EPRs(and possibly even the T-20) must be carefully investigated because of the large number of cycles envisioned per year. These range from 1000 in TFTR to 230,000 in the ANL-EPR. As shown in Figure 14, if the ASME design code were to be followed for the 538-650 °C case,such a variation in cycles implies limits on strains of 0.5 to 0.15%. The actual failure levels in irradiated steel at 600 °C appear to be 1.5 to 0.5%. Given all the complexities of vacuum joints,coolant channels, and penetrations in the blanket couples with temperature and damage gradients, it will be very difficult to insure that the strain per cycle in all the blanket structure is <1%.

Finally, with respect to the ductility of the structure, it is an open question as to whether the stainless steel first walls will be able to retain a reasonable amount of ductility (measured by uniform elongation) to withstand abnormal strains of as much as 1% during shutdown, start up, or other unforeseen transients. Figures 11 and 12 showed that above 20 dpa and temperatures of 428 to 650 °C, uniform elongation values of austenitic steels drop to below 1%. As the helium content and irradiation temperature is increased, this 1% level is reached earlier. The 600 °C operating temperature of the GA design may be particularly vulnerable to early failure due to this mechanism. While this appears to be one of the most limiting features of the EPR (and perhaps DPR) designs, more quantitative information on the ductility of metal irradiated under typical stress-temperature cycle condtions is required. The tolerable level of ductility needs to be clearly established by designers if one is going to be able to assess useful lifetimes for metallic components.

The final area of concern for neutron damage in the tokamak EPRs is the effect on the magnets. We have seen from Table 2 that the maximum displacement rates in the TF coils range from a low of 8 x 10^{-6} in the ANL design to 2 x 10^{-4} in the ORNL work. From Figure 9, we see that even after 10 year, the resistivity of the Cu in the ANL coil will roughly double due to radiation damage (assuming no intermediate anneals). On the other hand, the resisitivity in the GA copper toroidal field coil will double in a few years and that in the ORNL TF coil it will double in a few months. It appears that the ANL and GA designs can reasonably recover the damage by appropriate annealing every few years but the situation in the ORNL design is more serious. If the TF coil in the ORNL design were never annealed over the 10 year operation period, the resistivity of the Cu would increase by a factor of ~10! This would clearly be unacceptable in the event of an accident.

It appears that none of the EPR designs will produce enough damage in the NbTi superconductor to significantly alter its T_c and J_c values (see Figures 10a and 10b). However, the Nb_3Sn windings in the high field region of the ORNL magnet could suffer serious degradation in the T_c and J_c values. If this damage were never annealed out (the exact temperature required for this is unknown but it is probably 300-400°C for 95% recovery).The T_c value might be reduced by a factor of more than 10. Such degradations are probably not tolerable while still maintaining high reliability and therefore high temperature annealing or increased shielding is probably required.

The exposure level of 40 to 100 MRad per year to the organic thermal and electrical insulators does not seem to be a major problem.* The threshold for damage to mylar appears to be 8000 MRads, a factor of 4 above the anticipated 10 year exposure level in the ORNL design.

In summary, swelling in the EPR materials appears to be on the borderline between 'manageable' and a serious problem. The effect of stress on swelling will be extremely important here. The problem's introduced by irradiation creep are definitely quite serious, fatigue failures are definitely possible, and the high temperature in the GA blanket structure are **quite probable** and the high temperature in the GA blanket The Nb_3Sn and Cu stabilizer in the ORNL design appear to be vulnerable to damage.

4.3 Demonstration Power Reactors

Because of the extreme fluidity of the DPR designs at this time, it is not meaningful to analyze the operating parameters in great detail. However, there are a few qualitative remarks that can be made. On the positive side, the longer burn time will decrease the number of thermal cycles and therefore reduce the possibility of fatigue failure. We should also be able to better quantify the allowable stresses, strains and creep rates so that potential solutions can be tested before actual reactor operation.

On the negative side, the combination of higher wall loadings (1-3 MW/m^2) higher plant factors (approaching 70%) and longer plant lifetimes (up to 25y) will undoubtedly require frequent blanket replacement. It does not appear that integrated wall lifetimes of 35-50 MW/y/m^2 (250-500 DPA and 5000-10,000 appm He in steel) will be achieved in any presently known material operated at temperatures high enough to produce net electricity. Even if metals were operated at temperatures well below the onset of helium embrittlement, creep rupture lives of 200,000 hr at stresses approaching 100-200 MPa at 300-500°C will be **very**

* This conclusion only applies to the TF coils. The F coil in the GA design represents a much more severe problem. No specific exposure levels are known at this time but they are most likely to be in the 10^{10} Rad/year range for the present design.

difficult, if not impossible to achieve. In addition even if we could operate below the void formation temperature ($\sim 0.25\ T_m$) swelling due to high helium contents would limit the lifetimes to less than required.

The low Z liners also will face a higher frequency of replacement for the same reasons outlined above. However, such replacements, while difficult do not necessarily appear to be prohibitive both in terms of cost or time involved.

In summary, the materials problems in the DPR's will definitely force fusion reactor engineers into a design which places greater emphasis on accessibility than presently required by the EPR engineers.

5. Potential Facilities That Could be Used to Test Materials for Future Tokamak Reactors

It is not the purpose of this paper to review the irradiation testing field as that has been the topic of a recent conference.[40] There are 5 types of potential sources that could apply to the in-situ testing of Fe-Ni-Cr alloys for the experimental demonstration power reactors. These are (see Table 3);

A) The Rotating Target Neutron Source (RTNS) at the Lawrence Livermore Laboratory. An up graded version producing 2×10^{13} n $cm^{-2}s^{-1}$ (14 MeV) in a small ($\sim$1 cm^3) volume is currently under construction and scheduled for operation in 1978.[36]

B) The Intense Neutron Source (INS) at the Los Alamos Scientific Laboratory.[37] This source will produce up to 10^{14} n cm^{-2} s^{-1} (14 MeV) in a volume of few cubic centimeters and it should be operational in 1981.

C) A neutron stripping source such as the D-Li source proposed by workers at Brookhaven National Laboratory.[38] The neutron spectrum is rather broad and the testing volume can be stated as a function of the flux of total neutrons. Table 3 shows that $\sim$ 10cm^3 can be subjected to 10^{15} total neutrons $cm^{-2}s^{-1}$ whereas a volume of $\sim$300 cm^3 will experience fluxes above 10^{14} n cm^{-2} s^{-1} and $\sim$ 1000 cm^3 will see a flux of 5×10^{13} n cm^{-2} s^{-1}. This source is presently being designed and if funded in the U.S., may operate in 1983.

D) Because of the production of helium by thermal neutrons in Ni containing alloys one can consider the use of thermal test reactors. Our requirement that such facilities have sufficient testing space for in situ studies narrows the number of reactors down to those like the Advanced Test Reactor (ATR) [47] in Idaho Falls or the Oak Ridge Research REactor (ORR) [39] at the Oak Ridge National Laboratory. The flux levels here are not too meaningful because of different neutron spectra but on a displacement basis,such thermal reactors can produce damage at a rate three times higher than the RTNS and produce helium at $\sim$ 6 times the rate in RTNS. These values combined with a few hundred cm^3 of testing volume show the utility of such facilities. It would be relatively easy to modify such reactors for in situ testing in a year and such facilities could be operating in 1978.

E) A typical DT fusion reactor spectra could be provided by a driven (power and tritium consuming tokamak reactor similar to the Tokamak Engineering Test Reactor (TETR) described elsewhere.[34] The displacement rates are only twice that in thermal facilities and the helium production rates are comparable or slightly

Table 3

Possible Test Facilities to Provide Instrumented Data on Fe-Cr-Ni alloys for Fusion

Facility (Flux)	dpa/yr*	appm He/yr*	Instrumented Test Volume cm^3	PF%	Op Date	Ref
RTNS-UG (2×10^{13})	1.3	50	1	90	Jan '78	36
INS (10^{14})	6.5	250	3	80	Jan '81	37
D-Li (10^{15})	65	1000	10	80	Jan '83	38
D-Li (10^{14})	6.5	100	300	80	Jan '83	38
D-Li (5×10^{13})	3.3	50	1000	80	Jan '83	38
ORR	4.3	300**	300	70	Jan '78	39
TETR	11	200	2,000,000	70	Jan '87	34

*100% PF

** First Year

Table 4

Date When 1 Year Equivalent Insitu Data Could Be Produced (End of Year) By Projected Neutron Facilities For at Least One Specimen

	EPR		DPR	
	Max dpa	Max appm He	Max dpa	Max appm He
RTNS-UG	1980	1979	1990	1983
INS	1981	1981	1983	1982
D-Li (10^{15})	1983	1983	1983	1983
D-Li (10^{14})	1983	1983	1985	1986
D-Li (5×10^{13})	1984	1984	1988	1990
ORR	1978	1978	1982	1979
TETR	1987(a)	1987(a)	1988	1988

(a) system not applicable

less. However such a facility could provide a very large (a few million cm^3) volume for in situ testing. The drawback of this device is that even if the plasma physics and technology systems behave as anticipated it would be ~1987 before such a reactor could start to provide data.

When one combines the instantaneous damage rates with the anticipated plant factors and projected starting dates, he can calculate maximum cumulative damage levels that can be achieved. Such information is plotted in figure 23 and 24. As a point of reference we might compare this information with what might be required by designers of DPRs. It is reasonable that these designers would want information about at least one year of operation in a DPR and this information should be available by the start of construction of the reactor. Applying such criteria to the DPR we see that an average value between the GA and ORNL design would indicate that in situ tests of an austenitic steel up to 15 dpa and containing 300 appm He would be required by 1992. The interesting point about figures 23 and 24 is that all of the source considered for the DPRs and all except the TETR for EPRs, can produce the damage levels required by the proposed date of reactor construction. The appropriate displacement damage and helium production levels can be produced by the times listed in Table 4 for the two types of reactors.

However, just achieving the appropriate damage level is not sufficient for a successful materials test program. Many temperatures, environments stresses, alloy variations and back up samples must be tested in order to obtain a clear picture of the material's response to irradiation and to develop theoretical models. It is impossible at this time to establish a specific value for the number of samples that need to be tested and the total cumulative volume required for those tests. However, a recent estimate [34] for the PPR reactor arrived at approximately 5000 different samples that would need to be tested to provide a proper base to construct a DPR. At an average of 10 cm^3 per test (some post irradiation tests would require only a few cm^3 and some complex insitu tests might require 20-30 cm^3), this translates into a requirement for a damage-test volume product of about 10^6 dpa - cm^3 and ~ 2×10^7 appm He-cm^3 by 1991. Such a number could be off by a factor of 2 in either direction but it is doubtful whether it is off by as much as a factor of 10. The anticipated DPR requirements are plotted in figure 25 and 26 along with the cumulative damage-test volume product of the facilities listed in Table 3. The first obvious point in these figures is that none of the currently proposed facilities can provide the necessary data by 1991. The RTNS-UG is woefully inadequate in this respect and although such facilities cost only ~ 1-5% of the other facilities, even 20-100 such devices would not solve the problem. The next point is that a single INS is not adequate and more than 1000 would be required to produce the desired number of test specimens. The various D-Li combination are much better suited to provide this damage-volume ratio but again even under the best of circumstances, the product is a factor of 100 too low. The thermal reactors are not much better with regard to displacement damage but can provide data for four years before stripping sources are cosntructed. The situation with respect to helium is much better and the thermal reactors can provide information within a factor of 5 of that required (that is, if the samples

FIGURE 23

CUMULATIVE DISPLACEMENT DAMAGE IN STEEL FOR POTENTIAL CTR INSTRUMENTED IRRADIATION FACILITIES

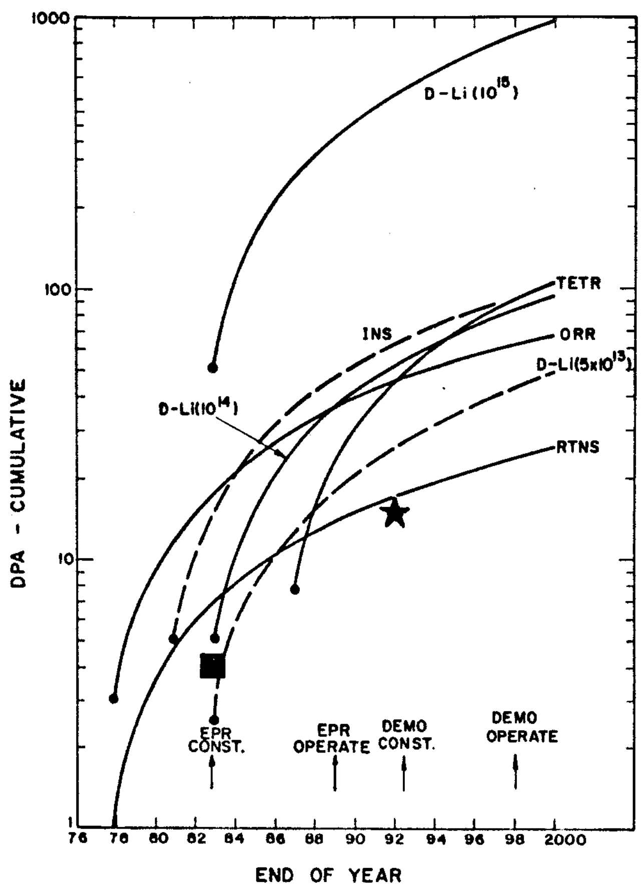


FIGURE 24

CUMULATIVE HELIUM PRODUCTION IN STEEL FOR POTENTIAL CTR INSTRUMENTED IRRADIATION FACILITIES

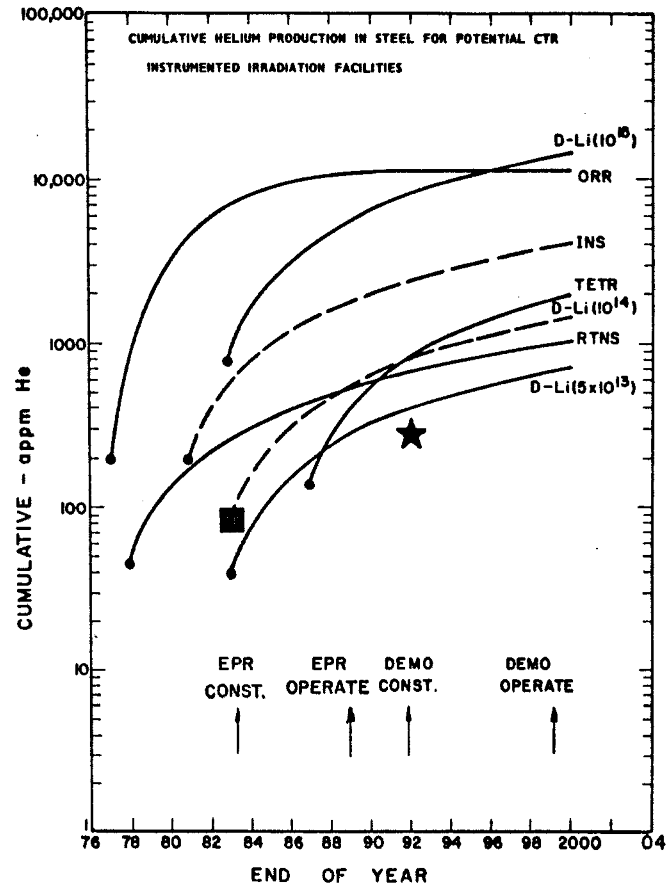


FIGURE 25

CUMULATIVE DAMAGE - TEST VOLUME THAT CAN BE PROVIDED BY POTENTIAL CTR INSTRUMENTED IRRADIATION FACILITIES

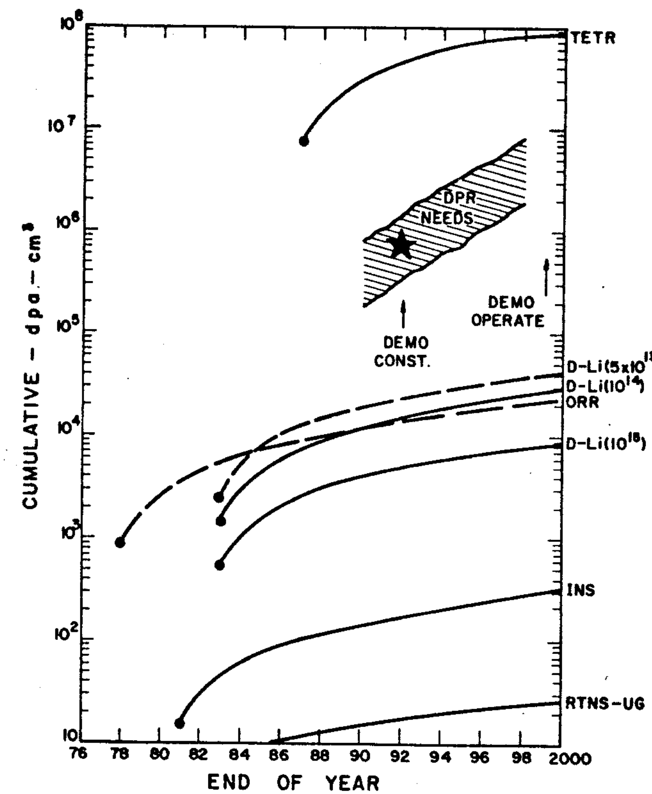
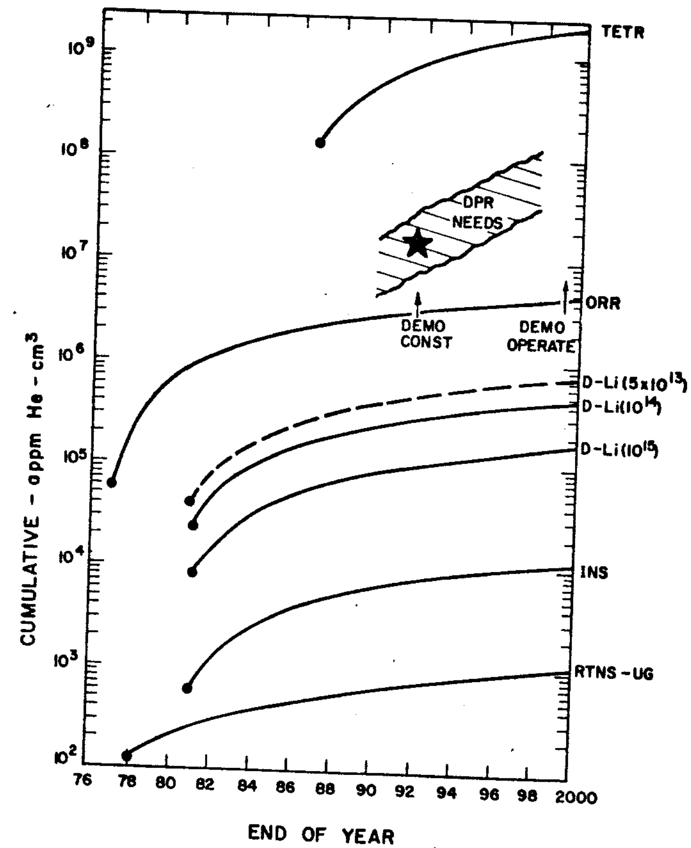


FIGURE 26

CUMULATIVE DAMAGE - TEST VOLUME THAT CAN BE PROVIDED BY POTENTIAL CTR INSTRUMENTED IRRADIATION FACILITIES



are removed every 4 years after the appropriate dpa damage is accumulated).

The only facility that could provide more than enough displacement and helium accumulation information is a DT plasma device like the TETR which operates at an average $1 MW/m^2$ over a year with a large testing volume. Such a device could be built with mid 1980 state of the art plasma physics and fusion technology if present research plans are met. The key to the advantage of such a device is the large ($\sim$ several million cm^3) volume of test area available. The cost of such a test facility may be in the 500 million dollar range so any judgement on this device versus multiple thermal fission, neutron stripping or solid target test facilities would have to balance the damage-test volume levels with cost and probabilities that such a device could be successfull built.

In summary, while current and proposed test facilities could produce a few specimens tested under realistic environments by the time the data is needed for a DPR, there is no facility except a large scale D-T reactor which can provide the appropriate number of samples. Further investigation will better define what is required in terms of a test matrix and what is ultimately possible from proposed test devices. Continued up dating of this information may provide the fusion community with more options to provide the necessary information.

6. Conclusion

The irradiation damage problems for the near term fusion devices are not expected to significantly effect their operation or the safety of such experiments. By the time the currently proposed experimental power reactors are built there will be several problems that could arise if these reactors are required to operate at high temperature. A great deal of testing and theoretical model development is required if such reactors are to operate in an efficient and safe manner. The situation is definately more critical for the DPRs and even if appropriate materials testing information can be provided, it is unlikely that any CTR first wall material will last the lifetime of the reactor. Therefore early fusion devices should be designed in a manner such that they can test remote maintenance techniques that will be required for later DPR and commercial reactors.

Acknowledgement

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