A Plan for Obtaining Timely Neutron Radiation Damage Information for Experimental Fusion Reactors

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Abstract

The need for irradiation damage information is examined with respect to the near term (1985 to 1997) D-T fusion reactors. The basic problem is that designers need a large variety of in situ data points on several materials approximately 4-5 years before the reactors are to operate. This means that the final design of the Experimental Power Reactors (EPRs) and Fusion Engineering Research Facilities (FERFs) must be completed in the FY 1981 to Fy 1984 period. Data on the effects of the uncollided 14 MeV neutron fluences of $7 \times 10^{20}\ cm^{-2}s^{-1}$, and perhaps 5 times that number in backscattered neutrons, on the properties of metals must be accumulated by that time. Both fast and thermal fission reactors appear to be unable to satisfy all of the data requirements because of temperature, spectrum or access limitations. Accelerator sources like the solid target D-T sources can provide the proper environment and radiation damage conditions but a very large number (~80) would be required by FY 78 in order to get the appropriate data by FY 81. Solid target D-Be sources could provide the proper access and radiation damage information if the reaction and displacement cross sections for 15-35 MeV neutrons were known. A reasonable solution appears to be the construction of a D-Be source and perhaps 10-20 D-T accelerator sources by the FY 78 period.

I. Introduction

The recent acceleration of the United States controlled thermonuclear reactor program[1] has presented materials scientists with their greatest challenge since the beginning of the nuclear fission program. Not only is the scientific community being asked to provide materials which can safely function in the harsh irradiation environment typical of a D-T plasma, but they must do so without the benefit of an extensive testing program. It is the object of this report to examine what information is needed, when it is needed and how that information might be obtained in time to have an effect on high power D-T fusion reactors of the 1980's. This analysis is not meant to be the last word on this complex and sometimes controversial topic, but it is hoped that it might simulate thoughts and ideas so that we all may be in a position to enjoy the benefits of fusion power by the early 21st century.

This report has four sections. First, the anticipated timetable of reactor operation is discussed with regard to its demands. Next the amount of information required for reasonable design decisions is discussed with respect to types of materials, temperatures, neutron fluences, environments, etc. This possible near term testing facilities are then examined with primary emphasis on accelerator type neutron sources using solid targets. Finally, a possible scenario is presented that could satisfy the severe requirements for materials radiation damage information in the 1980's.

II. Timetable of CTR Materials Requirements

The schedule proposed by the U.S.A.E.C. Division of Controlled Thermonuclear Research in January 1975 is given in Table 1 and Figure 1.[1,2] We have listed a few of the major parameters of the proposed devices such as:

plasma current,
fuel type,
magnet design,
tritium breeding requirements,
neutron wall loading,
plant factor,
and, disposition of the energy generated.

An estimate of the timetable for preliminary design, funding requests, final design, construction, and operation is given in Table 2. We will describe only briefly the purpose of each device here and the reader is referred elsewhere for more descriptive material.[3,4] A list of the abbreviations used in this report is given in Appendix A.

The main purpose of the PLT is to provide plasma physics information on circular toroidal plasmas while the PDX will help us understand the removal of impurity atoms by divertors. The Doublet-III will provide information on non-circular plasmas. All of these devices will "burn" hydrogen and therefore do not have neutron radiation damage problems or major heat removal problems.

The TFTR will be the first device to use a D-T fuel.[4] The instantaneous wall loading will be ~0.1 MW/m^2 and only ~4000 total "shots" will be fired over the lifetime of the reactor (approximately 1000 shots per year). Each shot will last only a few seconds so that the integrated 14 MeV neutron fluence is at most $\sim 8 \times 10^{15} n/cm^2$ per year.

Table 1

Summary of U.S.A.E.C.-DCTR Proposed Fusion Devices in the 1980-2000 Period-Tokamaks

Device	Current MA	Fuel	Magnets	Breeding	Wall Loading MW/m^2	Plant Factor	Power Disposition
PLT	1-2	H_2	Cu	No	-	-	-
PDX	?	H_2	Cu	No	-	-	-
D-III	1-2	H_2	Cu	No	-	-	-
TFTR	2-3	D-T	Cu	demonstrate	~0.1	10^{-4}	dump
EPR-I	~5	D-T	S/C [a]	?	0.1-0.2	0.1-0.5	electricity?
FERF-I	?	D-T	S/C	? limited	~1	~0.5	dump?
EPR-II [b]	5-10	D-T	S/C	Yes	~0.5	~0.5	electricity
FERF-II [b]	?	D-T	S/C	probably some	~2	~0.7	dump?
DPR [b]	10-15	D-T	S/C	Yes	~1	~0.7	electricity

[a] S/C = superconducting

[b] estimated by authors

Table 2

Proposed Plan to a Tokamak Demonstration Power Plant

Device	Preliminary Design	Funding Request	Fiscal Year Final Design	Start Construction	Operation
PLT		1971	1971-2	1972	1976
PDX	1971-4	1974	1974	1975	1988
D-III	1972-4	1974	1974-5	1975	1978
TFTR	1974-5	1975	1976	1977	1981
EPR-I	1974-7	1978	1979	1980	1986
FERF-I	1975-9	1979	1980	1981	1986
EPR-II*	1979-82	1982	1983	1984	1990
FERF-II*	1979-82	1982	1983-4	1984	1990
DPR *	1984-88	1988	1989	1990	1997

*author estimates

Reactors of the EPR series are to operate in 1985 and in 1990.[5-7] They will provide knowledge of plasma physics at higher currents while incorporating superconducting magnets, perfecting heating techniques, fueling devices, etc. The wall loading in the EPR's will be kept relatively low (~0.1-0.2 MW/m^2) and it is not clear whether the power generated will be converted to electricity. Therefore, the walls may run cold (~50-200°C) or quite hot (high enough to generate usable steam, ~200-500°C). The second EPR will most likely operate at higher wall loadings, higher plant factors and higher temperatures. Significant materials problems may be encountered in EPR-II because of the higher wall loadings and higher temperature.

The two materials test reactors must include not only a high wall loading but high plant factor as well. It is not clear whay type of reactor will be used (mirror, pinch or tokamak) but integrated flucences of ~0.5 MW/m^2 should be attainable in a year. The first FERF will probably not be required to generate electricity and therefore the walls may run cold. We will not speculate on its requirements for the second FERF at this time.

Finally, the DPR will certainly operate with high wall loadings, high plant factors and high temperatures. Its successful operation depends on how well the previous test facilities have performed. It is expected that this device must demonstrate the <u>ability</u> to generate electricity economically which also implies that it must have high plant factors.

III. Required Information on Materials

A. General Considerations

Before discussing the details of what data are needed at what temperature for what material, it may be worthwhile to outline some of the most important assumptions we have made in this work.

1. We have assumed the introduction schedule of Figure 1 and Table 2.
2. We have assumed the reactors would operate at the conditions stated in Table I starting at the beginning of the fiscal year given for operation (Table 2).
3. The uncollided 14 MeV neutron flux is assumed to accumulate at a rate equal to $4.43 \times 10^{13} nsec^{-1} cm^{-2}$ x (wall loading in MW/m^2) x (plant factor)
4. It is assumed that the last time the reactor designers can make meaningful changes in the device is at the end of the final design period and the beginning of construction.
5. It is also assumed that the designers would want to have data equivalent to at least one full year of reactor operation before they freeze the design.
6. The reactor designers need information on both the primary and secondary structural material choices for range of temperatures, neutron fluences, stresses and coolant environments which are typical of the reactor operating conditions.

There are several possible measures of the neutron fluence requirements, such as:

A. Uncollided 14 MeV neutron fluences

B. Total neutron fluence including back scattered neutrons

C. Displacement damage

or D. Helium generation.

The relation between A and B depends on the blanket configuration but the backscattered flux is ~5 times the uncollided 14 MeV neutron current to the first wall. On the other hand, the energy of the back scattered neutrons is much lower than 14 MeV so that they will produce relatively less radiation damage. Therefore, the number of displacements per atom (dpa's) is probably a better measure of the damage in the material because it takes into account the neutron spectrum as well as the total number of neutrons. Finally, the rate at which helium atoms are produced is a sensitive measure of the neutron flux above the (n,α) reactions. This is important for neutron sources whose spectrum extends significantly above 14 MeV. The rate of helium production strongly influences the high temperature ductility of a metal.

Let us now first examine the damage accumulated in the first walls of the various reactors as a function of time and then estimate the number of data points required for one reactor system, FERF-I.

B. Specific Damage Considerations

We will discuss only those materials which are most likely to be used in the near term tokamak reactors. These are listed in Table 3. It appears that some form of austenitic stainless steel would be used as primary alloy for the reactors up to the end of the century while some Al alloys may be attractive in low temperature-low radiation damage systems because of the relatively low induced radioactivity. Vanadium alloys are attractive at higher temperature in moderate radiation environments and Mo alloys may be used in systems toward the end of the century when both high temperatures and reasonable radiation damage resistance are important. It should also be noted that significant amounts of carbon might be present in the devices operated after 1985 so that

Table 3

Summary of Most Likely First Wall Materials
to be Used for Near Term Fusion Reactors

Reactor	Primary	Secondary	Tertiary
TFTR	Steel	Al alloy	-
EPR-I [a]	Steel	Al alloy	
FERF-I [a]	Steel	V alloy	Al alloy
EPR-II [a]	Steel	V alloy	Al alloy
FERF-II [a]	Steel	V alloy	Mo alloy
DPR [a]	Steel	V alloy	Mo alloy
Commercial Power [a] Reactor	Mo alloy	V alloy	Steel

(a) It is quite possible that various forms of carbon might be used in these machines to protect the plasma from contamination or to protect the first wall by degrading the neutron spectrum.

damage effects on carbon should be considered.

We have listed the various damage parameters in Table 4 for the devices and materials outlined in Table 3. The dpa and helium levels were calculated using the rates per MW/m^2 from reference 8 and are summarized in Table 5. The back scattered spectra were assumed to be those of a metal-liquid Li-graphite reflected blanket. Such an approximation is probably good to 20% which is acceptable at this stage of analysis. Let us now discuss the results in terms of the three methods of analysis (A, C and D).

The accumulated uncollided 14 MeV neutron fluence is presented in Figure 2 with the latest times at which the designers must have the designated fluence data in hand to design the reactor. This figure shows that significant radiation damage accumulation does not occur in any fusion device until ~FY-86. The accumulated uncollided 14 MeV neutron fluence is ~$7 \times 10^{20} n/cm^2$ in FY-86 for FERF-I. This is 5 to 50 times higher than will be accumulated in the DPR-I high and low designs.* The FERF-I continues to accumulate a 14 MeV neutron flux at the rate of $7 \times 10^{20} ncm^{-2} year^{-1}$ until its total fluence is surpassed by FERF-II in FY-1994. This occurs at ~4×10^{22} neutrons/cm^2. FERF-II will accumulate 14 MeV neutron fluences equal to the expected lifetime fluences in the DPR before the DPR is operated. On the other hand, this information may not be useful to the DPR because, according to the schedule in Table 2, it is obtained 4 years after final design freeze has occurred.

It is interesting to note that practically all of the design data for the near term machines are needed in the 4 year period between FY-1980 and FY-1984. Data corresponding to an uncollided 14 MeV neutron fluence of ~10^{20} to $2 \times 10^{21} n/cm^2$ are required at that time.

* See Table 4 for definition of EPR ranges

Table 4

Summary of Damage Parameters in Near Term Tokamak Devices

Reactor	Date of Design Freeze	Initiation of Damage	Uncollided 14 MeV Flux n cm^2 yr^{-1}	Displacement damage dpa/year					Helium Production Rate appm/year				
				Steel	Al	V	Mo	C	Steel	Al	V	Mo	C
TFTR	EOFY-76 (a)	SOFY-83 (b)	1.4×10^{16}	10^{-4}	2×10^{-4}	-	-		2×10^{-3}	4×10^{-3}	-	-	-
EPR-I (c) (high)	EOFY-80	SOFY-86	1.4×10^{20}	0.98	1.7	-	-		20	41	-	-	-
EPR-I (c) (low)	EOFY-80	SOFY-86	1.4×10^{19}	0.098	0.17	-	-		2.0	4.1	-	-	-
FERF-I	EOFY-81	SOFY-86	7×10^{20}	4.9	8.5	5.9	-	4.7	101	206	28	-	1195
EPR-II	EOFY-83	SOFY-90	3.5×10^{20}	2.	4.3	2.9	-	2.4	51	103	14	-	600
FERF-II	EOFY-84	SOFY-90	2×10^{21}	14.0	-	16.8	-	13.5	290	-	81	-	3410
DPR	EOFY-89	SOFY-97	9.8×10^{20}	6.9	-	8.2	5.8	6.6	142	-	40	33	1670
CPR													
LINS (90%)			3.4×10^{20}	0.76	0.65	0.77	0.60	0.43	26	41	7.8	6.1	290
D + Be (90%)													

(a) EOFY = End of Fiscal Year

(b) SOFY = Start of Fiscal Year

(c) High represents 0.2 MW/m^2 at 50% Plant Factor

Low represents 0.1 MW/m^2 at 10% Plant Factor

Table 5

Summary of Typical Damage Rates in CTR Materials Due to Backscattered Neutron Spectra[8]

	Rate per year per MW/m^2 (14 MeV neutrons)	
Material	dpa	appm Helium
Stainless Steel	9.8	200
Al	17	410
V	12	57
Mo	8.2	47
C	9.5	2400

Accumulated uncollided 14 MeV neutron fluence in various experimental tokamak fusion reactors. The asterisks give the time at which data at indicated fluence are needed for design of reactor.

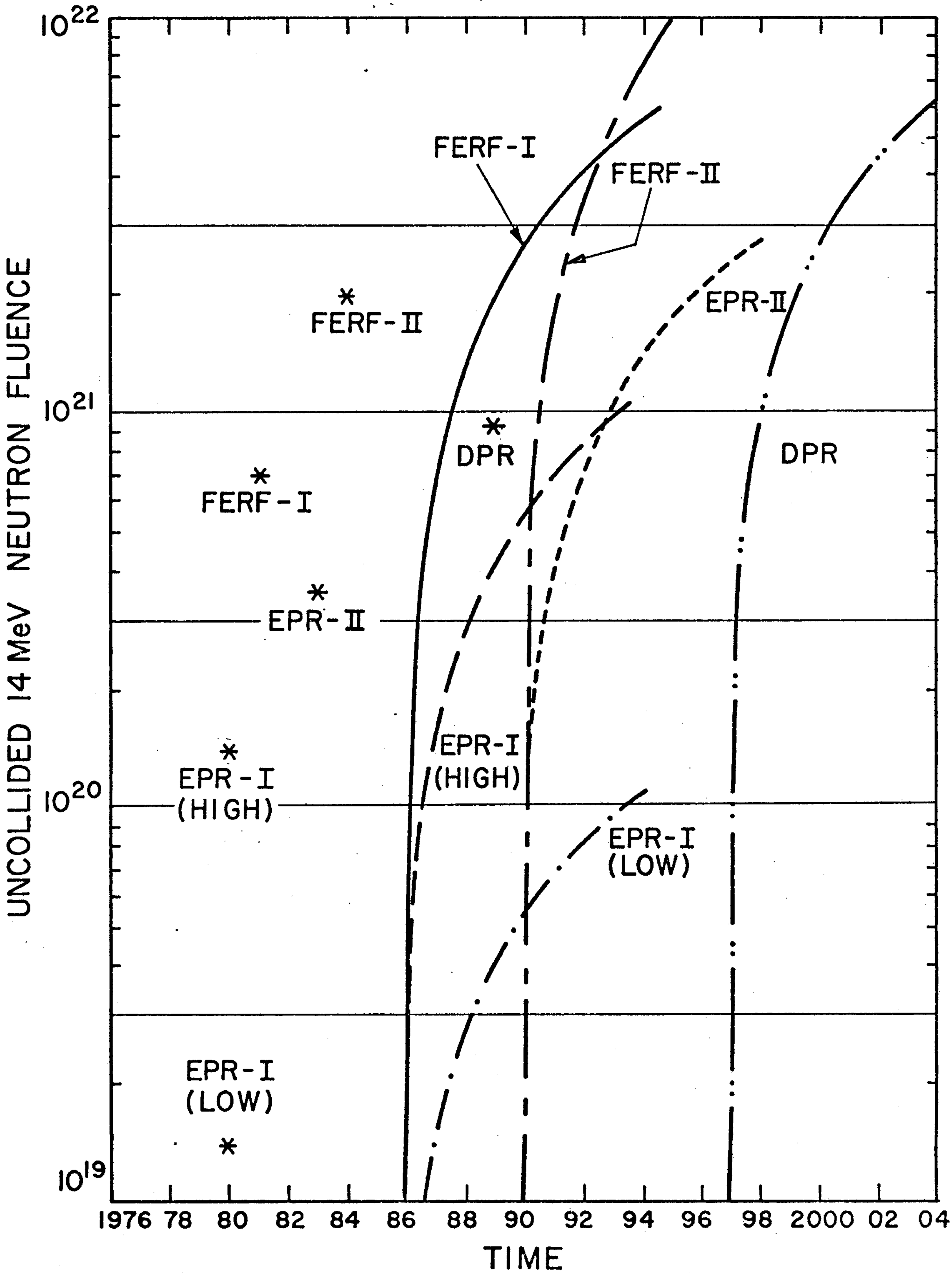


FIGURE 2

Accumulated displacement damage in steel in various experimental tokamak fusion reactors. The asterisks give the time at which data at indicated damage are needed for design of reactor.

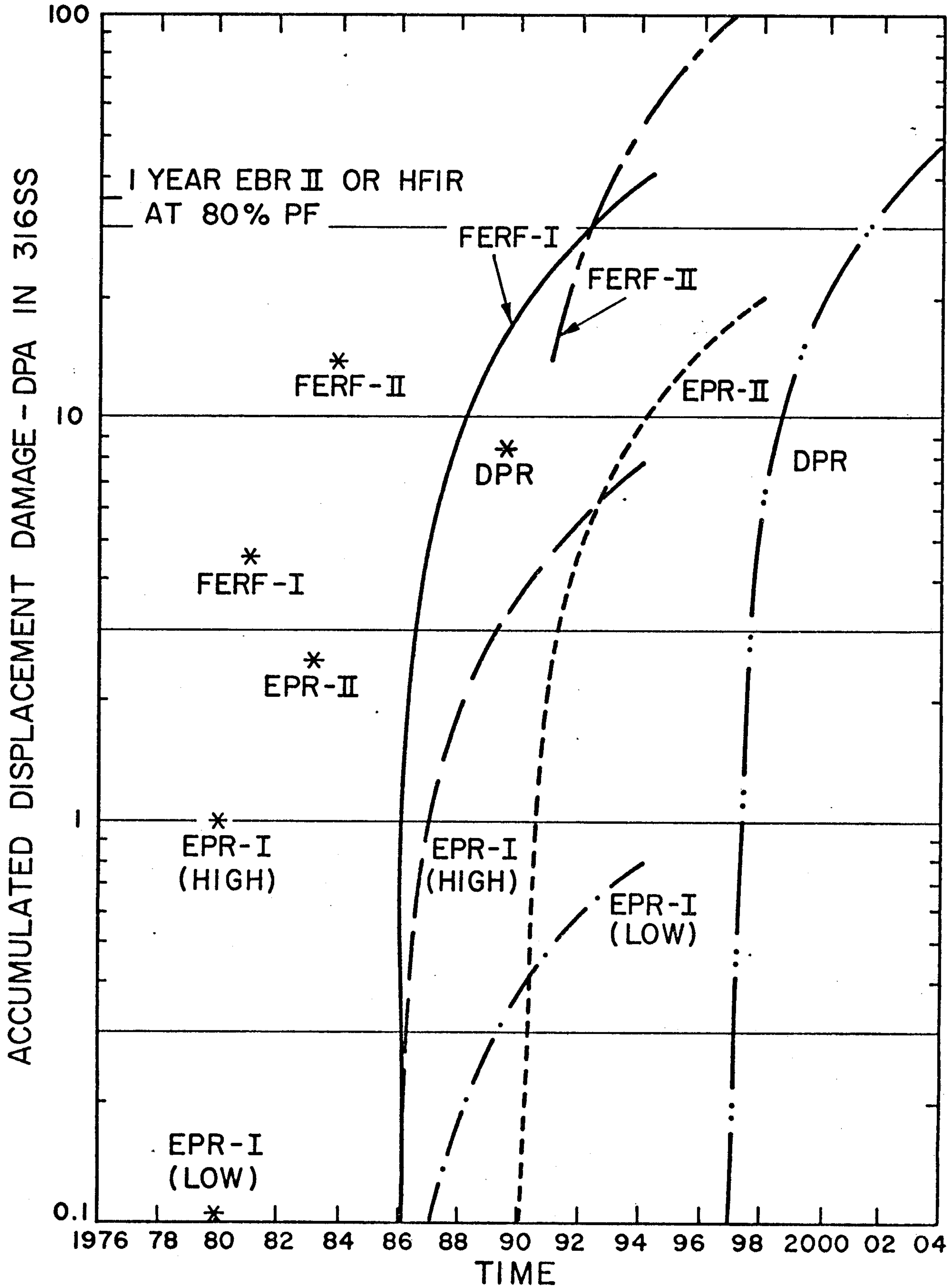


FIGURE 3

The accumulation of displacement data for steel are given in Figure 3. The results would be almost the same for any other material. The important points are:

1. Significant dpa damage (~1 dpa) in CTR devices will not be accumulated until the start of FY-86.
2. Damage levels in the EPR-I vary from 0.8 to 8 dpa after an assumed 8 year lifetime.
3. The damage is highest in FERF-I up to FY-92, where it is approximately 30 dpa.
4. The decision points for all the near term devices occur from FY-1980-FY 1984 and data are required in the 1-13 dpa range.

A comparison on the material needs based on the helium production is also given in Figures 4a and 4b. Figure 4a is for 316 stainless steel and Figure 4b is for aluminum. The 1980-1984 decision points require from 20-280 appm helium concentrations in 316 SS and 40-500 appm in Al to simulate one years exposure. For comparison, the helium production rate in 316 SS in EBR-II is ~4 appm per year at an 80% plant factor and ~10 appm per year for Al. It is also important to note that the steel in the DPR could contain ~700 appm He at the end of life. Such a level will not be reached in <u>true</u> CTR environment until ~FY 1993 and then simultaneously in FERF-I and FERF-II.

Similar figures could be drawn for V, Mo or C from Table 4 and the conclusions would be approximately the same, namely, that the period between 1980 to 1984 is critical for the development of many high power fusion devices. Furthermore, the period of 1975-1980 is most important for gathering data on the proposed materials for these devices. The most critical requirement is to obtain sufficient data on materials for the FERF-I decision in FY-81 (only

Accumulated Helium in Steel

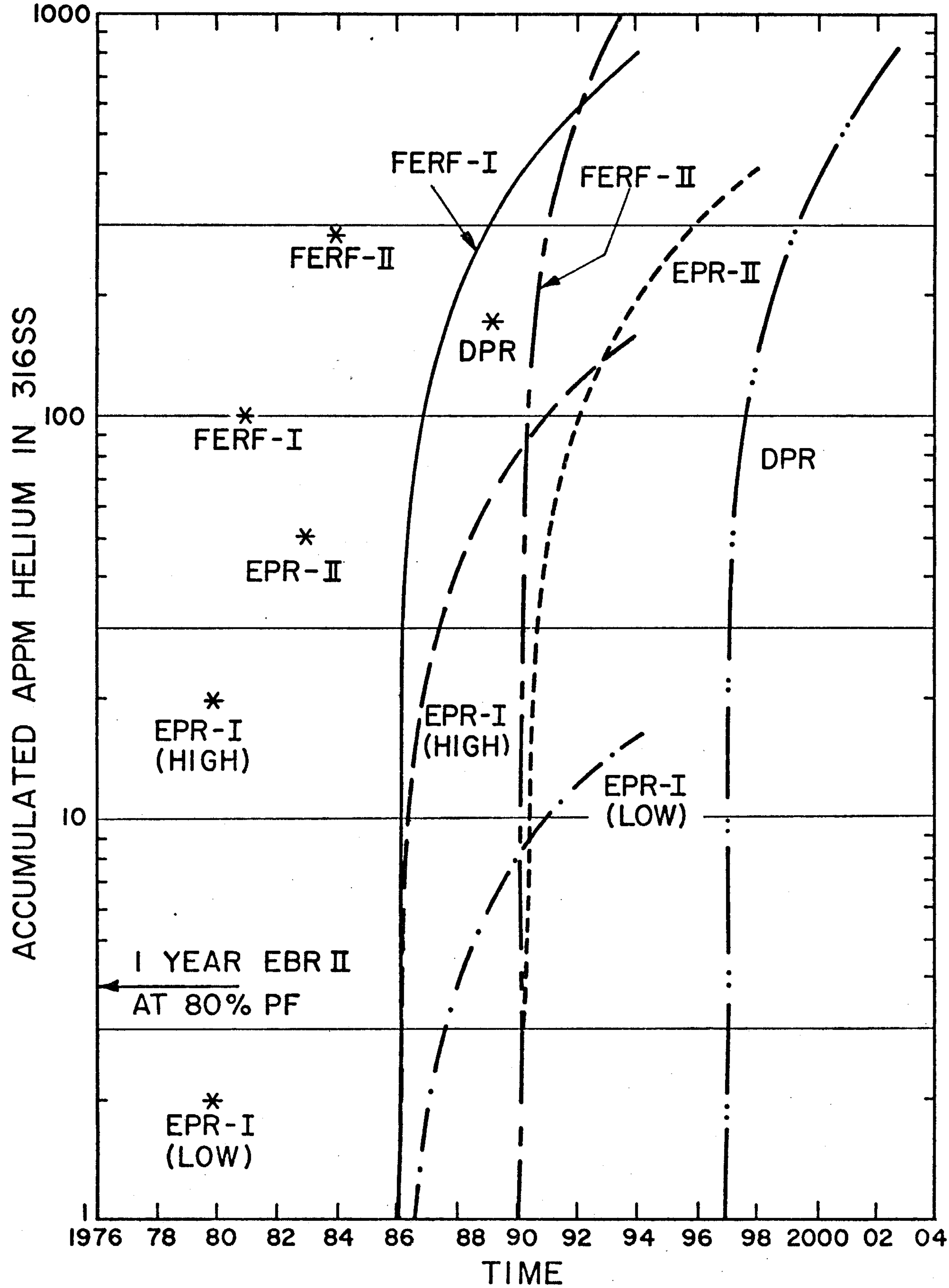
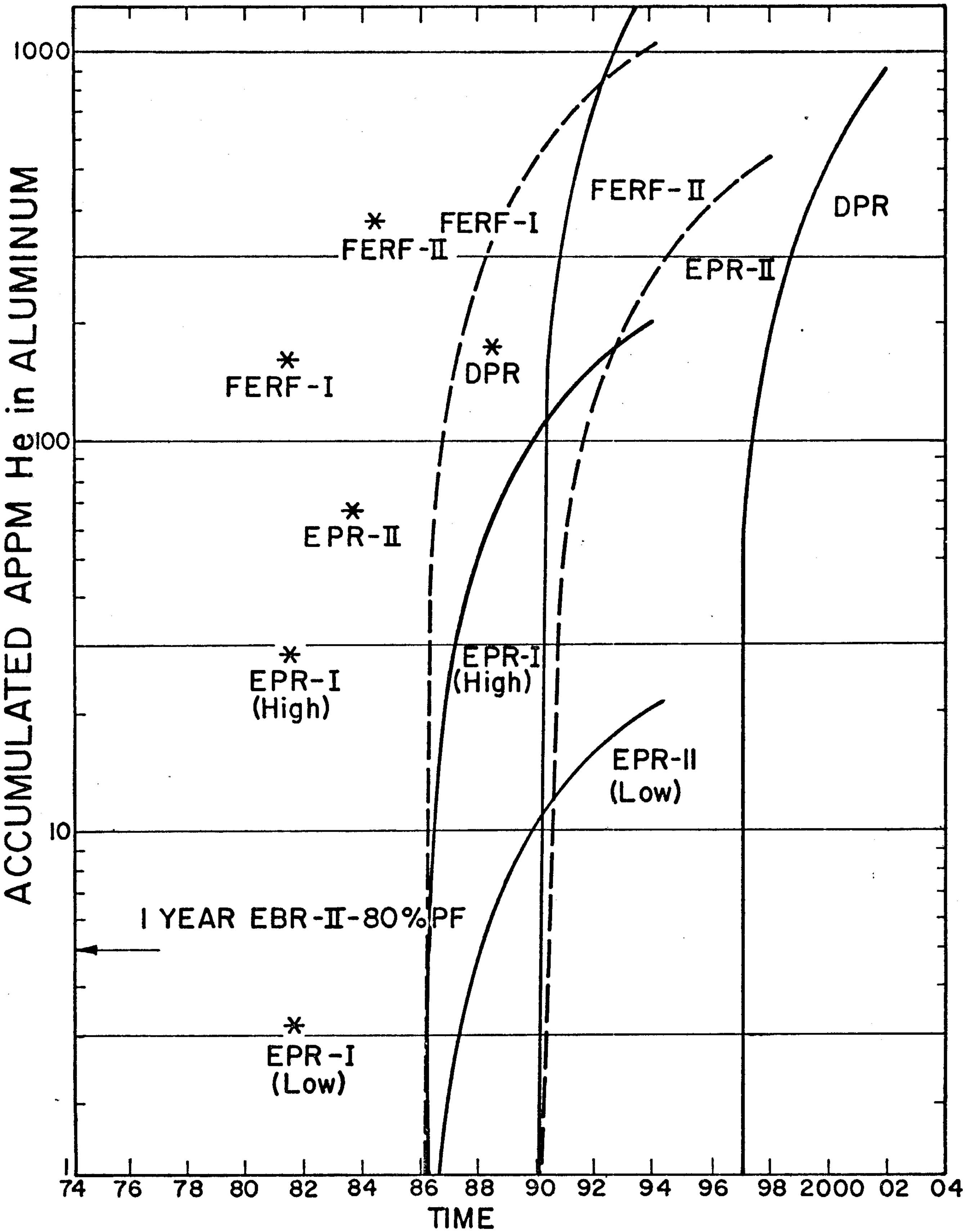


FIGURE 4a



Accumulated helium in aluminum in various experimental tokamak fusion reactors. The asterisks give the time at which data at indicated helium concentrations are needed.

FIGURE 4b

5 years from the time of this writing). Let us now focus our attention on only that reactor and attempt to determine not only the level of damage information required, but how many data points are needed at various temperatures, stresses, and coolant environments.

C. A Preliminary Assessment of the Number of Data Points Required for FERF-I First Walls

This type of an assessment requires a large number of assumptions on the alloy, temperature, stress level, and coolant material which might be used for such a reactor. In order to simplify the discussion we make the following assumptions, which may have to be revised later.

1. FERF-I will not produce electricity.
2. FERF-I will not produce tritium.
3. The coolant will be water.
4. The first wall temperature will be between 50 and 200°C.
5. The 14 MeV neutron wall loading will be 1 MW/m^2.
6. The duty cycle will be 50%.
7. The bulk of the reactor will have a lifetime of 5-8 years although special modules will be designed to be changed yearly.
8. The only materials considered will be 316 SS because of experience gained in fast breeder reactors and Al because of its low activation.

At present, the most important materials problems for the Fusion Engineering Reactor Facility are (a) Failure due to excessive creep, or steady state deformation of the wall material from the coolant pressure, (B) Failure caused by the cyclic nature of the thermal stresses (failure from thermal fatigue could occur either through growth of fatigue cracks to the point that failure occurred in response to nominal coolant pressure loading, or in response to localized loads which may be encountered during maintenance operations, etc.), (C) Loss of ductility as a result of helium and displacement induced embrittlement, and subsequent failure during local loading as a result of maintenance or other operational events, and (D) Failure caused by excessive swelling, either through loss of required dimensional control of through excessive stress developed as a result of swelling gradients.

The experimental programs required for the design of a CTR FERF first wall will depend upon the wall temperature and material selected. The effect of these parameters on the required swelling studies can be readily assessed from LMFBR experience. The void formation process has been found to be important in the temperature range $0.3 \leq T/T_m \leq 0.5$, (where T_m is the melting point). The higher helium production rates associated with the CTR environment may extend the upper temperature limit which is influenced by nucleation, but probably not the lower limit which is thought to be associated with vacancy migration and emission rates. Consequently, swelling behavior studies are important if the wall material operates at temperatures in excess of 0.3 T_m. This means that the homologous temperatures of irradiation of 50 to 200°C are 0.18 to 0.26 for 316 SS, and 0.35 to 0.5 for Al. Such low irradiation temperatures preclude significant swelling in steel, but are in the appropriate range for swelling in Al. A minimal test program is required for 316 SS to determine the extent of distortion and dimensional changes to be expected from point defect or cluster buildup and the associated lattice parameter changes. On the basis of these discussions, the test matrix in Table 6 may be considered a reasonable minimum for the design of a test reactor.

Density measurements of sufficient accuracy could be performed on sheet samples of the following approximate dimensions: 0.25 mm thick x 2.5 mm wide x 10 mm long. The effects of constant and cyclic applied stress on swelling may also be of concern, but these perturbing effects can be explored through post examination of fatigue and creep sepcimens to be described later.

The mechanical properties test matrix can be developed most conveniently from the use of deformation-mechanism maps which display the fields of stress and temperature in which a particular mechanism of plastic flow is dominant. Ashby[9] has described the development and use of such maps for materials in which athermal dislocation glide and various thermally activated creep mechanisms dominate the deformation process. Holmes and Lovell[10,11] have extended the use of these maps to include the irradiation creep deformation mechanism. The deformation mechanism map for annealed Type 316 stainless steel[11] is shown in Figure 5 in terms of temperature and strain rate. The map is shown in terms of stress and temperature in Figure 6. In Figure 6, the map was simplified by considering both dislocation creep and Coble creep as components of the thermally activated flow mechanism. Also indicated on Figure 6 is an elastic region which indicates the stress and temperature regions over which the strain rate is anticipated to be less than 5×10^{-8}/hr. This corresponds to a total plastic deformation of less than 0.05% after two years operation at 50% plant factor. The dashed line on the figure indicates the boundary between the elastic region and the thermally activated flow regions in the absence of irradiation creep. The size of the irradiation creep field will depend upon the displacement rate and perhaps to some extent upon the nature of the displacement. This particular deformation map was developed assuming a displacement rate of 10^{-6}dpa/sec., which is typical of LMFBR's and 2-3 times that expected for most CTR applications.

If a stainless steel wall is used, stress values may be as high as 15 to 20 ksi (100-140 MPa) if the cyclic thermal gradient stresses associated with the heat flux through the wall are included. This stress and temperature range is indicated in the shaded area on the deformation map in Figure 6. In addition to these normal operating loads, one must allow for localized

TABLE 6

SWELLING STUDY TEST MATRIX TO DETERMINE THE MAGNITUDE OF SWELLING IN STEEL AND ALUMINUM TEST SPECIMENS

Fluence	Temperature (°C)		
(14 MeV neutrons/cm^2)	50	125	200
2×10^{20}	XO	XO	XO
4×10^{20}	XO	XO	XO
5.5×10^{20}	X	X	X
7×10^{20}	XO	XO	XO

X Density (2 each)

O Microstructure (3 each)

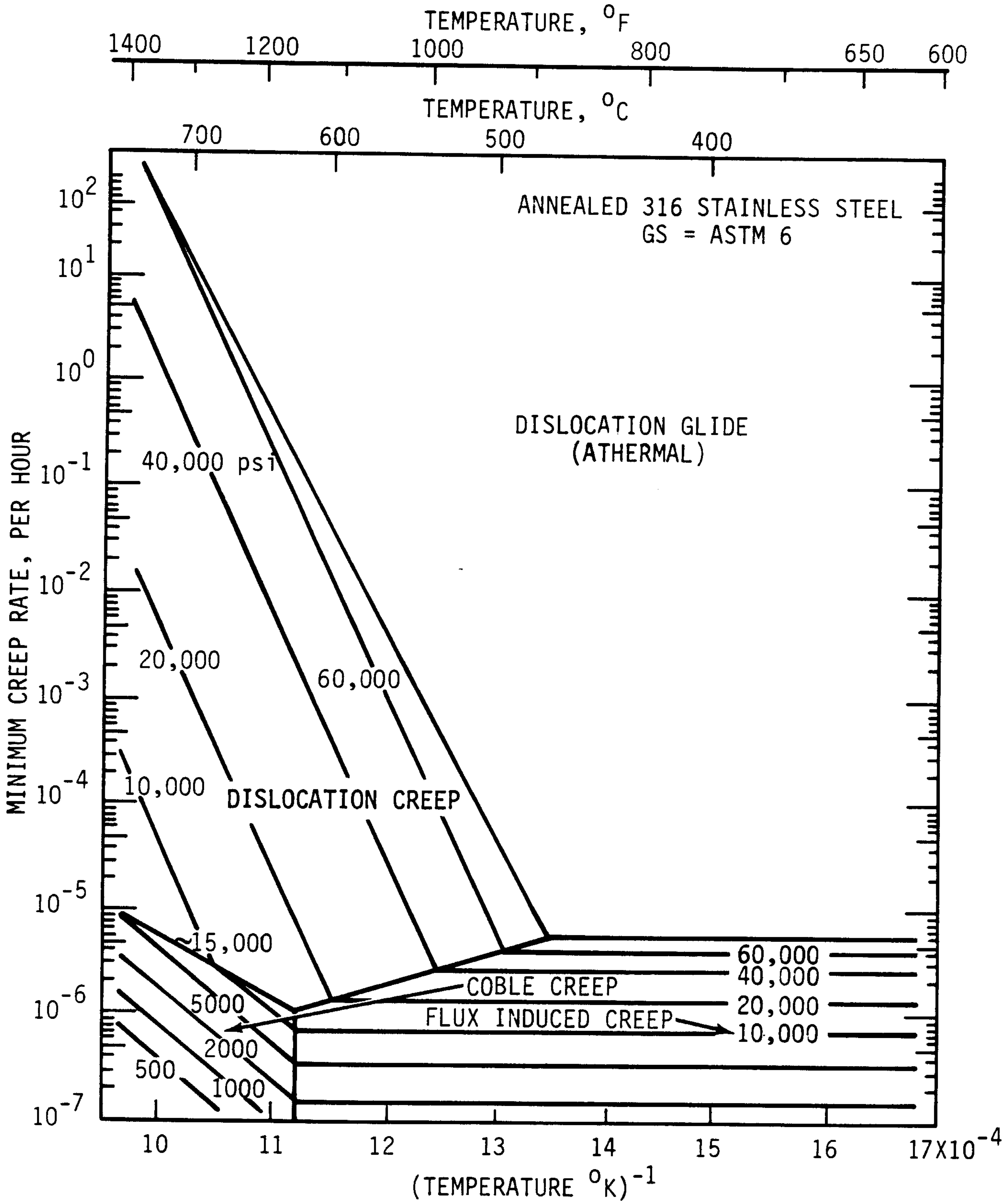


FIGURE 5
Deformation mechanism map for annealed 316 stainless steel

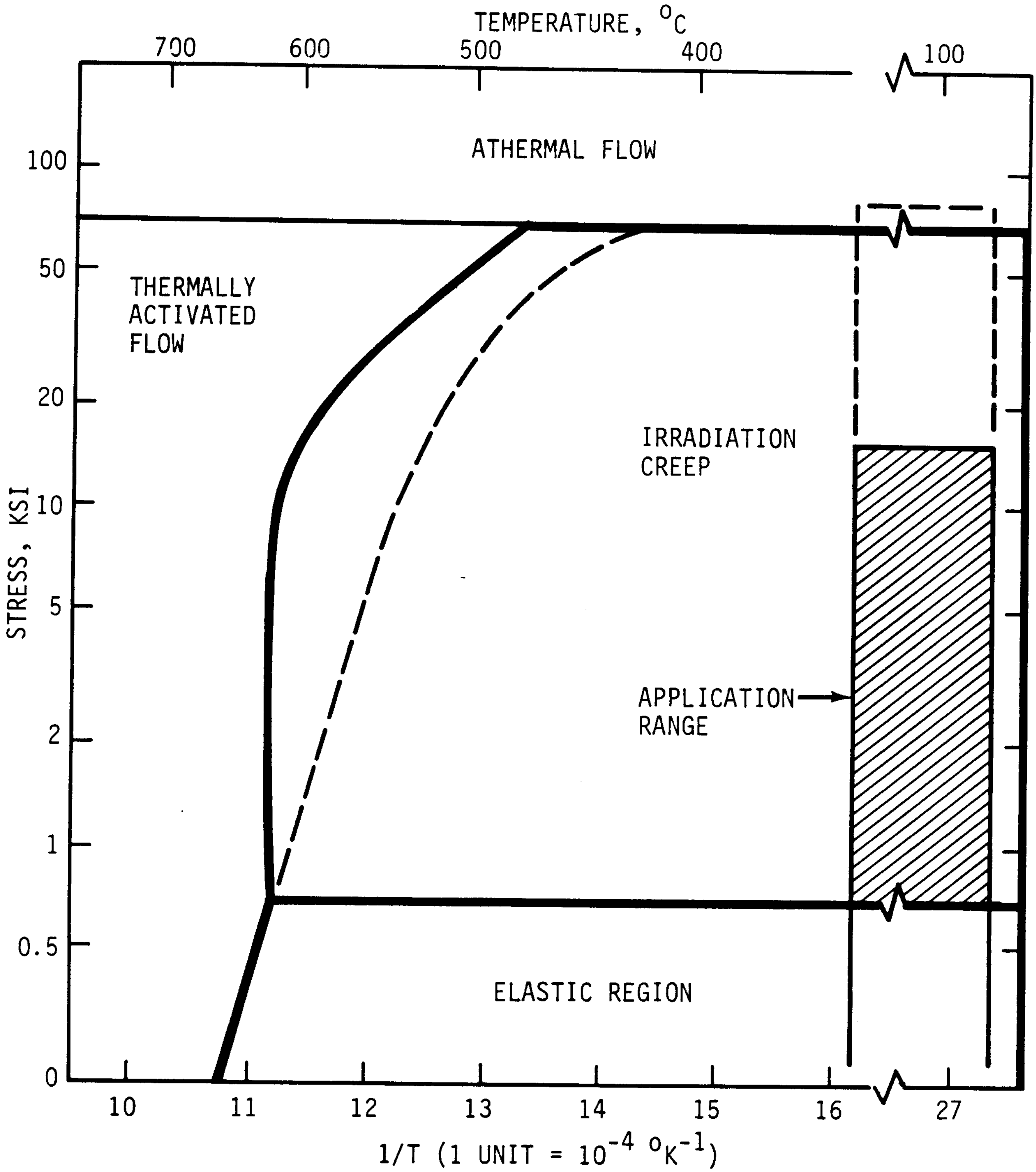


FIGURE 6
Simplified deformation mechanism map for annealed 316 SS. Dislocation creep and Coble creep are included in a thermally activated flow.

overstressing caused by effects such as long range thermal expansion effects, or unexpected incidents during maintenance or handling operations. This type of loading will be referred to as transient loading. The major mechanisms of transient loading will be secondary or strain limited so that ductility will be the primary property of interest. The transient loading stress and temperature ranges are indicated in the unshaded area enclosed by dashed lines in Figure 6. Inspection of Figure 6 indicates that if the first wall of FERF is austenitic stainless steel, several phenomena are not of major concern. While it may be important to future commercial reactors operating at higher temperatures, helium embirttlement is not expected to be of particular significance in the test reactor since it is important only in the high temperature thermally activated deformation range. Kangalaski and Shober(12) have extensively tested Type 347 stainless steel after irradiation in a thermal reactor at fluences as high as $2.1 \times 10^{22} n/cm^2$ ($E < 1$ MeV).* These materials were irradiated in water at 50°C and experienced displacement and helium generation levels even higher than expected in the FERF after 2 years operation. Tensile testing at high temperatures (750°C), in the thermally activated flow range resulted in nil ductility, whereas ductility values of several percent were observed after tests in the athermal range. More recently, Bloom and Wiffen(13) have investigated tensile properties in austenitic steels after irradiation to fluences as high as $8.7 \times 10^{22} n/cm^2$ at temperatures from 500-700°C. In this study the ductility in the athermal range was reduced to 0.5%, but the displacement damage (~50 dpa) and helium levels (~4000 appm) are considerably above those expected in FERF. In general, for deformation in the athermal range, low temperature irradiation increase the yield strength and reduces work hardening

* This corresponds to roughly 10 dpa.

to the point that the yield strength approaches the ultimate strength. Material with these properties exhibits good ductility under secondary loading conditions, but because of the lack of work hardening, is unstable where substantial strain is produced in sections with varying thickness. On the basis of experience with thermal reactor irradiations and of the anticipated adequate ductility, only minimal testing is required for the post-irradiation tensile behavior of austenitic stainless steel. These tests are indicated in Table 7. A 0.25 mm thick miniature sheet tensile specimen would be sufficient for these tests.

Thermal fatigue and/or thermal ratcheting also appear to be of little concern because under the assumed operating conditions, the material is expected to be either in the irradiation creep or elastic deformation regime. Thermal stresses which will develop during neutron bombardment will be relaxed by irradiation creep processes. However, Figure 6 shows that the material will be in the elastic regime so that negligible deformation will occur when the machine is shut down and the stress gradient is reversed. Consequently, as the number of cycles and total exposure time increases, the system will tend toward nil stress gradient during the burn and equal to the reverse of the peak thermal stress during the "down" portion of the cycle. Therefore, the maximum plastic and cyclic deformation expected over the life of the component will be negligible and no significant cyclic fatigue or ratcheting effect is expected. The confirmation of these expectations could be obtained through four in situ cyclic load tests indicated in Table 8 to generate an "S-N"* curve. In general, one may expect the fatigue behavior to be sensitive to the adjacent medium (coolant, vacuum, etc.). In these tests the water environment will not be necessary as the surface on the water side will tend to be in compression.

* S-N = Number of cycles (N) to produce failure at a given stress (S).

TABLE 7

TENSILE TEST MATRIX FOR AUSTENITIC STAINLESS STEEL FIRST WALL

Fluence (14 MeV neutrons/cm^2)	Temperature (°C) 50	200
3.5×10^{20}	X	X
7×10^{20}	X	X

X = 3 tests each at 2 strain rates with test temperature equal to irradiation temperature - 3 tests at room temperature - 1 strain rate. (9 tests for each X)

TABLE 8

CYCLIC FATIGUE TESTING FOR AUSTENITIC STAINLESS STEEL WALL

A. In Situ S-N Curves

Stress Range (ksi)*	Cycle Rate (cycles/day)	Temperature
20	5	200
30	5	200
40	5	200
50	5	200

B. Fatigue Crack Growth Specimens

4 Specimens - Irr.@200°C to 7 x 10^{20} n/cm^2(14 MeV)

*1 ksi = 6.895 MPa

Four post-irradiation fatigue crack growth specimens should also be included to permit calculation of wall life for a certain initial flow distribution.

A deformation mechanism of importance to the first wall performance, and requiring quantitative study, is irradiation creep. However, because of the relatively small temperature dependence and linear stress dependence, irradiation creep requires relatively few tests. The relatively low levels of average stress and the minimal dimensional stability requirements of the first wall permit a design which is not very sensitive to the irradiation creep. Table 9 shows a test matrix compatible with this reasoning. Self-loaded specimens which are periodically removed for creep measurement could be used for these tests.

Aluminum may be selected for a first wall material because of the low radioactivity and because of its high thermal conductivity and resultant low thermal stresses. A deformation map for commercial aluminum alloys was not available but an approximate one was obtained by modifying the stress and temperature scales with the modulus and melting temperatures of Al and stainless steel. The resulting deformation map and anticipated service conditions are shown in Figure 7. The assumed operational stress for aluminum is assumed to be considerably lower, primarily because of the much lower thermal stresses. As may be seen from Figure 7, the required tests for aluminum are much more extensive than for stainless steel, because the thermally activated flow processes are significant in the first wall performance. The tensile test matrix in Table 10 provides the detail necessary to characterize embrittlement.

Inspection of Figure 7 leads to the conclusion that the cyclic effects will be very complex. The inside surface of the wall is expected to creep in the compression direction by irradiation creep while the reactor is on,

TABLE 9

IRRADIATION CREEP TESTING FOR AUSTENITIC STAINLESS STEEL WALL

Stress Level (ksi)	Temperature °C 50	125	200
5	X	-	X
10	X	-	X
15	X	-	X
30	X	-	X

Max. Fluence = $7 \times 10^{20} n/cm^2$ (14 MeV)

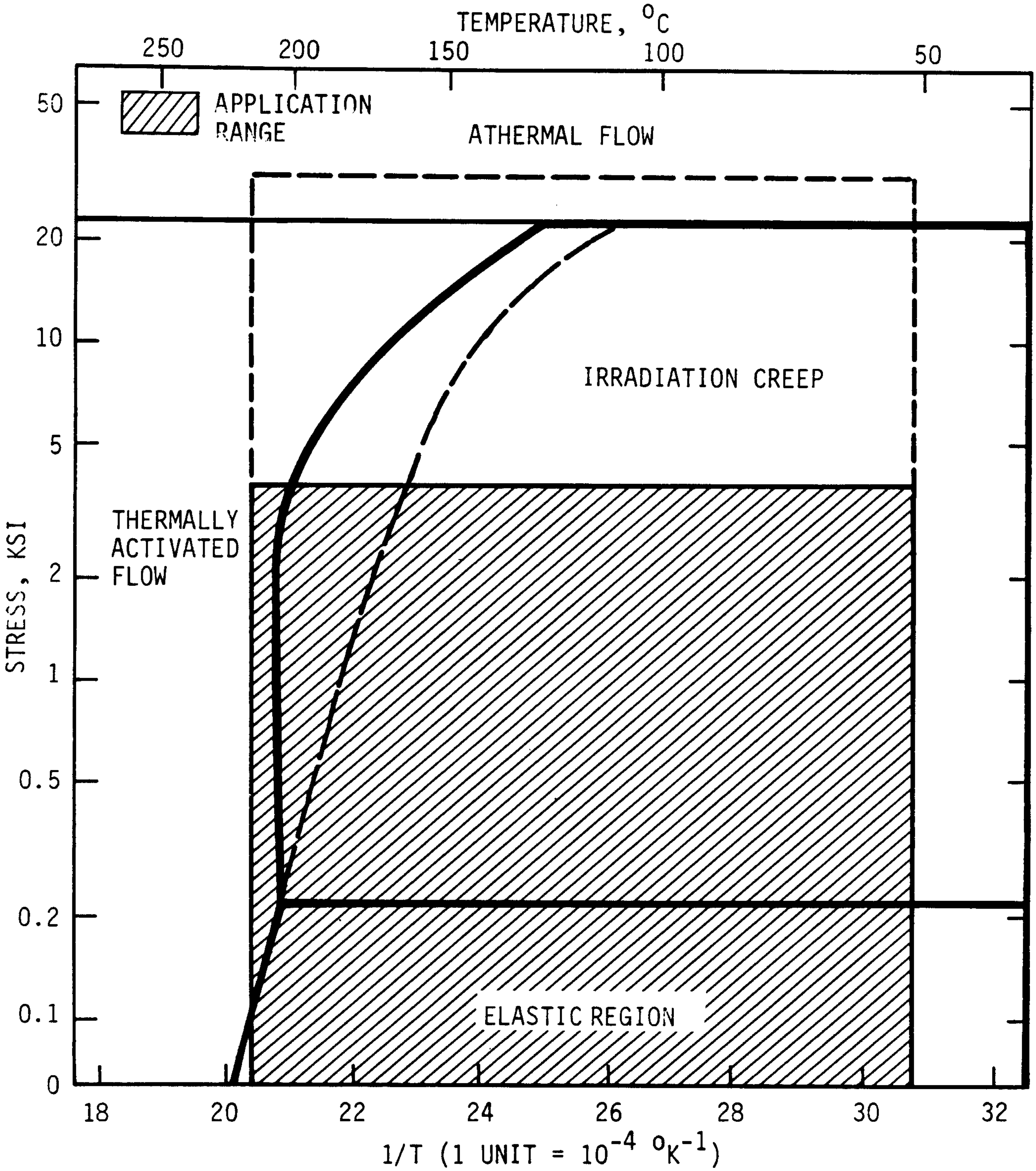


FIGURE 7

Simplified deformation for aluminum alloy and to creep in the tension direction by thermal creep just after the reactor is turned off. Obviously, this fatigue process cannot be adequately studied by post-irradiation techniques and in situ techniques will be necessary. The cyclic behavior of the aluminum should be studied at several temperatures, because it may be sensitive to temperature in the anticipated operating regime.

Since both the inside and outside surfaces of the wall may experience either tension and compression a minimal amount of testing will be required in both water and vacuum. The maximum exposure for the in situ tests will be the same or slightly less, than that for one year of operation. Some of the cyclic stressed specimens will not fail. The effects of cyclic loading on the transient behavior (i.e., structural integrity) could be determined by tensile testing of the unfailed specimens. A measure of fatigue crack growth rates will be necessary to make quantitative estimates of wall life given the surface quality of the wall material. These tests must be performed in situ in order to obtain the proper combination of deformation mechanism at the crack tip. The test matrix indicated in Table 11 shows these requirements.

In situ creep tests, as shown in Table 12, must be more detailed than for austenitic stainless steel because the operating conditions encompass both irradiation and thermally activated creep. Consequently, the creep behavior is expected to be sensitive to the temperature in this range.

A summary of the number of irradiation tests requires to qualify (or verify) the use of steel and aluminum as first wall materials for FERF is given in Table 13. The requirements can be divided into two areas: In situ tests and post-irradiation tests.

The next chapter will discuss the time required to perform these experiments and the neutron sources that could be used for the irradiations.

TABLE 10

TENSILE TESTS REQUIRED TO CHARACTERIZE THE BEHAVIOR OF Al FOR FIRST WALL APPLICATION IN TEST REACTOR

Fluence	Temperature °C			
(14 MeV neutrons/cm^2)	50	100	150	200
3.5×10^{20}	X	*	*	*
7×10^{20}	X	*	*	*

X 3 tests each of 2 strain rates - test temperature = irradiation temperature

* 3 tests each at 4 strain rates - test temperature = irradiation temperature and 3 tests each at 1 strain rate - tests performed at room temperature

TABLE 11

CYCLIC TESTS FOR ALUMINUM

A. In Situ. S-N Curves

Stress Range (ksi)	Cycle Rate (cycles/day)	Temperature Range °C 50	150	200
0.5	5	X	X	X
1.0	5	X	X	X
2.0	5	*	*	*
4.0	5	X	X	X

B. In Situ Crack Propagation Studies

Stress Range	Temperature Range °C 50	200
Variable	*	*

C. Fatigue Crack Growth Specimens

4 specimens each irradiated @ 50°C, 150°C, 200°C

Irradiation in available space provided by machines for parts A and B.

TABLE 12

IN REACTOR CREEP TESTING FOR ALUMINUM

Stress Level (ksi)	50°C	100°C	150°C	200°C
1	X	X	X	X
2	X	X	X	X
4	X	X	X	X
8	X	X	X	X

Max. Fluence = 7×10^{20} n/cm^2 (14 MeV)

TABLE 13

SUMMARY OF MATERIALS IRRADIATION TESTS REQUIRED FOR VERIFICATION THAT STAINLESS STEEL AND ALUMINUM CAN BE USED IN THE HYPOTHETICAL FERF-I DESIGN

Parameter Studied	Number of Tests: Steel	Aluminum	Total
Swelling[a]	51	51	102
Tensile Test[a]	36	108	144
Cyclic Fatigue (Insitu S-N)[b]	4	12	16
(Insitu Crack Propagation)[b]	–	2	2
(Crack Growth)[a]	4	12	16
Irradiation Creep[b]	8	16	24
Total	103	201	304
Total In Situ Tests	12	30	42
Total Post Irradiation Tests	91	171	262
Total	103	201	304

[a] Post Irradiation Tests

[b] In situ Tests

IV. Current and Near Term Neutron Sources Which May Be Applied to CTR Materials

A. Introduction

The first point to recognize in any simulation study is that it is important not only to duplicate the neutron flux and fluences, but the neutron spectrum as well. For a typical D-T fusion reactor, this means that we must not only produce a high flux of 14 MeV neutrons, but one would like to also simulate the flux of degraded neutrons which is approximately 5 times higher than the uncollided 14 MeV flux. This degraded (or back scattered) spectra is similar to that in a fission reactor.

We will discuss only three possible neutron sources for obtaining data on fusion materials and we will limit our remarks here to the sample case we outlined in Chapter III for steel and aluminum in FERF-I. Two of the facilities considered are accelerator sources, a solid target D-T source and a solid target D-Be source. The third facility is the Fast Flux Test Facility, FFTF.(14) It is tempting to consider a thermal neutron test reactor but the only one available with high enough neutron flux is the High Flux Isotope Reactor (HFIR).(15) Unfortunately, it is not built for instrumented (in-situ) testing of components in the reactor core, and post-irradiation testing does not always indicate the damage state present during irradiation. A brief description of these sources follows.

B. Fast Flux Test Facility

A detailed summary of this reactor design is given in reference 14. Table 14 presents a summary of the information pertinent to the present study.

There are four noteworthy features of this facility -

1. The minimum temperature for testing is 315°C (the reactor inlet coolant temperature);

2. A large sample volume is available ($10^5 cm^3$) in the useful ($>0.2\ \phi_{max}$) flux area;

3. There are almost no neutrons above 10 MeV;

4. The flux of neutrons of energy greater than 1 MeV is high.

The first feature makes this facility not particularly useful for testing materials at low temperatures typical of the FERF-I reactor. The third feature prevents the achievement of helium and hydrogen gas production rates characteristic of a CTR spectrum although the fourth feature results in a much higher rate of displacement damage than in FERF-I. Finally, there will be a rather large volume of sample testing space available ($\sim 10^5 cm^3$). This space is presently earmarked for LMFBR materials test programs and it may be difficult to get enough dedicated to CTR work until the late 1970's. Duplication of such a facility is difficult because of the cost (~400 million dollars) and time required for construction (~7 years).

C. Solid Target D-T Neutron Source

The neutron source that produces at present the highest D-T neutron flux over extended periods is the Rotating Target Neutron Source (RTNS) at Lawrence Livermore Laboratory (LLL).[16] In this source about 20 mA of 400-keV atomic deuterons bombard a Ti layer in which tritium is absorbed. The target is 22 cm in diameter and rotates at 1100 rpm. The source strength from a fresh target is about 2.6×10^{11} neutrons/mC incident on the target. The available flux depends on the deuteron beam diameter, which determines the size of the target spot, and on the distance between source and sample. For TEM and surface studies of radiation damage, samples between 5 and 10 mm in diameter and 0.1 mm thick have usually been used. In an 80-hour irradiation fluences between 1.5 and $3 \times 10^{17} cm^{-2}$ are usually obtained on such samples.

TABLE 14

SUMMARY OF IRRADIATION TEST FACILITIES WHICH MAY BE USED TO OBTAIN IRRADIATION DAMAGE DATA ON POTENTIAL CTR MATERIALS

	FFTF	LINS	D-Be	FERF-I
Operation (approximate)	1978	1977	1978	1985
Spectra	fission	'pure' 14 MeV	0-30 MeV	0-14 MeV
Access	difficult	easy	easy	difficult
T min -°C	315	(unlimited)	(unlimited)	50
Useful Sample Volume-cm^3	10^5	0.8	1.6	10^6
Useful Maximum Neutron Flux-Total $cm^{-2} sec^{-1}$	2×10^{15}	10^{13}	3×10^{15}	3×10^{14}
Flux for E > 1 MeV	10^{15}	10^{13}	3×10^{15}	2×10^{14}
Flux for E >10 MeV	Neg.	10^{13}	2.3×10^{15}	10^{14}
Flux for E >15 MeV	Neg.	none	1.4×10^{15}	Neg.

A similar facility capable of producing higher intensities is being designed at LLL. The new facility (Livermore Intense Neutron Source)[17] is expected to produce initially a source strength of $10^{13} s^{-1}$ and larger $4 \times 10^{13} s^{-1}$. Although higher source strength may eventually be achieved, this would require further higher source strength may eventually be achieved, this would require further development work. The available neutron flux is a sensitive function of sample to source distance. For a source strength of $4 \times 10^{13} s^{-1}$ and a 1-cm diameter deuteron beam, a flux of $10^{13} cm^{-2} s^{-1}$ could be obtained at a sample to source distance of 0.4 cm. Figure 8 shows the expected variation of the neutron flux over a sample. The flux is axially symmetric around the axis of abscissae. The contour marked 0.3 describes the surface within which the flux is greater than $0.3 \times 4 \times 10^{13} cm^{-2} s^{-1}$. We have listed some of the pertinent information on the LINS in Table 14 for comparison with other sources. The useful volume has been considered to be a contour which is ~2 mm by 20 mm long and 20 mm wide or $0.8 cm^3$.

All of the discussion thus far has been about the uncollided 14 MeV neutron current to a sample. For radiation damage studies one would like to expose materials to accelerator produced neutrons that have a spectrum that is as similar as possible to the spectrum at the inner wall of the reactor. One might think that this could be done easily by surrounding the target of a D-T generator with a mockup blanket. Such an arrangement does not actually produce the desired spectrum for two reasons. The accelerator produced neutrons have a spectrum which depends on the energy of the bombarding charged particles, on the thickness of the target, and on the direction of observation with respect to the incident beam. If one employs the neutrons emitted near the forward direction, as is usually done, the neutron energy is significantly higher than that for thermonuclear neutrons. The cross sections for hydrogen and helium production may be substantially different at the two energies and therefore the effect of the accelerator produced neutrons may not be exactly the same as that of the primary thermonuclear neutrons.

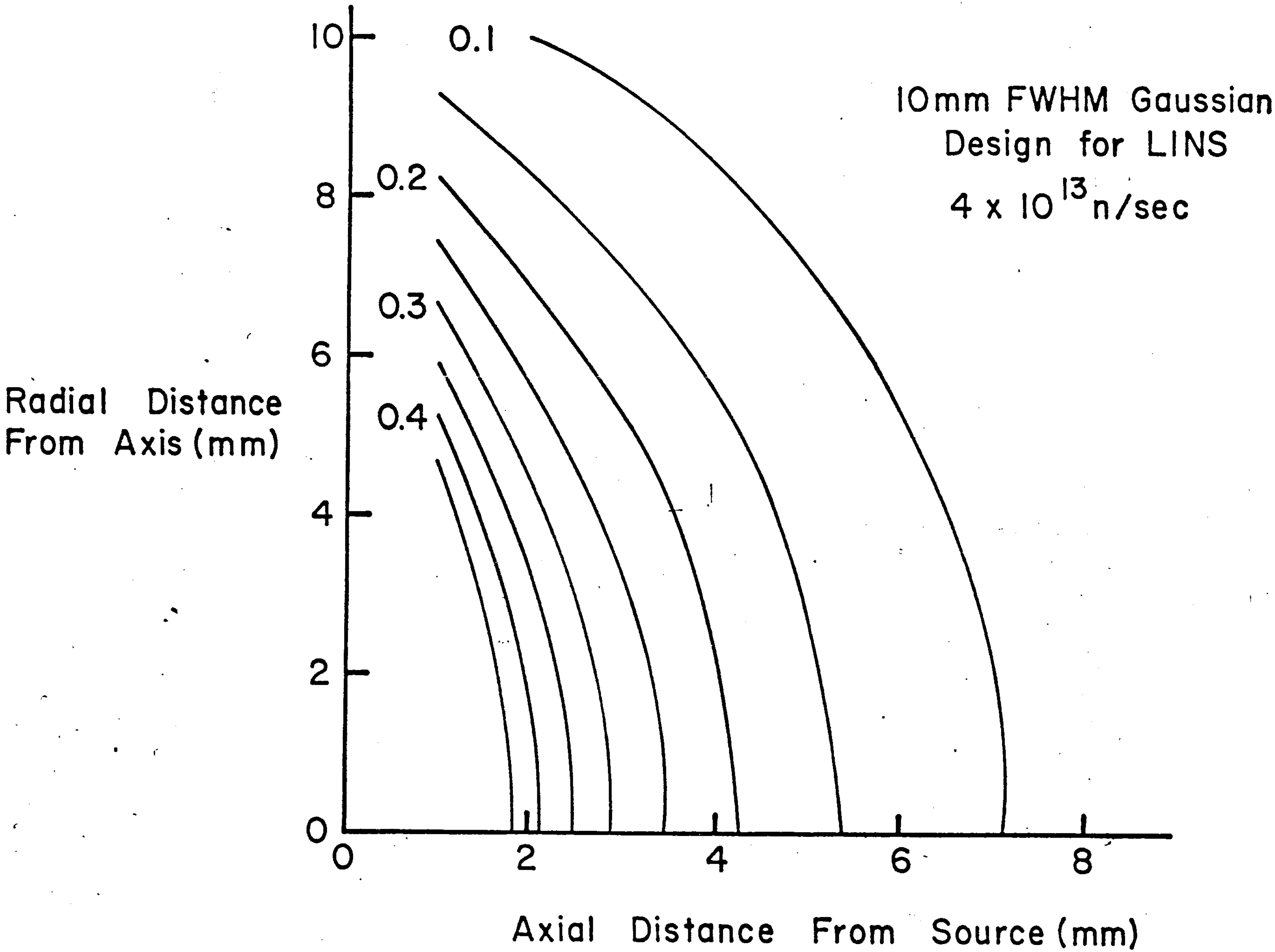
In order to obtain adequate neutron flux from the accelerator, samples have to be placed as close to the target as possible. This makes it almost impossible to modify the primary spectrum by placing blanket material near the target. The effective source of the neutrons that have collisions in the blanket is of the order of a mean free path inside the blanket. This is of the order of 5 cm, while the sample may be 0.5 cm from the target. Taking into account the inverse square decrease in flux with distance, one can see that the blanket material will not modify the effective primary spectrum substantially.

D. Be(d,n) Source

Since accelerator produced D-T neutrons do not have the exact spectrum desired, one may wish to consider other neutron sources that have a spectrum not too different from that in a fusion reactor. The most promising of these is the reaction of deuterons with beryllium. This reaction has been used for a long time for producing high intensities of energetic neutrons. For a deuteron energy of 33 MeV, the neutrons emitted in the forward direction from a thick target hace a continuous spectrum that peaks near 14 MeV and extends from zero to about 30 MeV (see Figure 9.)[18] This differs from the reactor spectrum even more than the accelerator produced D-T neutron spectrum so that even larger corrections for the spectral difference have to be applied to the data. On the other hand, it is very much easier to obtain high neutron fluxes from the D + Be reaction than from the D +T reaction.

In order to apply a correction for the spectral difference between the in FERF and those from d+Be nuclear cross sections have to be measured,

FIGURE 8



Neutron flux expected from Livermore Intense Neutron Source for a neutron source strength of $4 \times 10^{13} s^{-1}$ and 1 cm diameter deuteron beam. The flux is cylindrically symmetric around axis of abscissae. The contour marked 0.3 gives the distance within which the neutron flux exceeds $0.3 \times 4 \times 10^{13} cm^{-2} s^{-1}$.

preferably over the entire neutron energy range up to the maximum neutron energy in the d + Be spectrum. Since only relatively poor neutron energy resolution is needed, such measurements are quite straight forward. He production can be measured directly for the d + Be spectrum, and such an integral measurement can be compared with measurements as a function of neutron energy. One would expect the He production cross section to be higher for d + Be neutrons than for 14 MeV neutrons so that for the purpose of planning experiments with d + Be neutrons, the assumption of equal cross sections is conservative.

Although it might be possible to obtain fluxes as high as $10^{14} cm^{-2} s^{-1}$ from D-T sources, this will require the development of new technology, and it seems very unlikely that fluxes greater than $10^{14} cm^{-2} s^{-1}$ can be obtained from accelerator type D-T sources. It does appear, however, that without new technology, fluxes as high as $10^{15} cm^{-2} s^{-1}$ could be obtained from the D + Be reaction. In order to have a spectrum that peaks around 14 MeV, the deuteron energy should be around 30 MeV if neutrons produced in the forward direction from a thick Be target are to be used. At this bombarding energy the yield of neutrons in the forward direction is about $3 \times 10^{14} mC^{-1} sr^{-1}$.

On the basis of the experience with the linear accelerator that serves as an injector for LAMPF it appears possible, without substantial development, to accelerate deuteron beams up to about 10 mA to 30 MeV. At 1 cm from the target, the flux would then be about $3 \times 10^{15} cm^{-2} s^{-1}$ of which more than $2 \times 10^{15} cm^{-2} s^{-1}$ have energies above 10 MeV.

The deuteron beam would deposit 300 kW in the target, and this requires effective cooling. Because of the high melting point of Be and its fairly good heat conductivity, the cooling problem does not appear difficult. If the target is thick enough to stop the deuteron beam (0.4 cm) and if the metallic

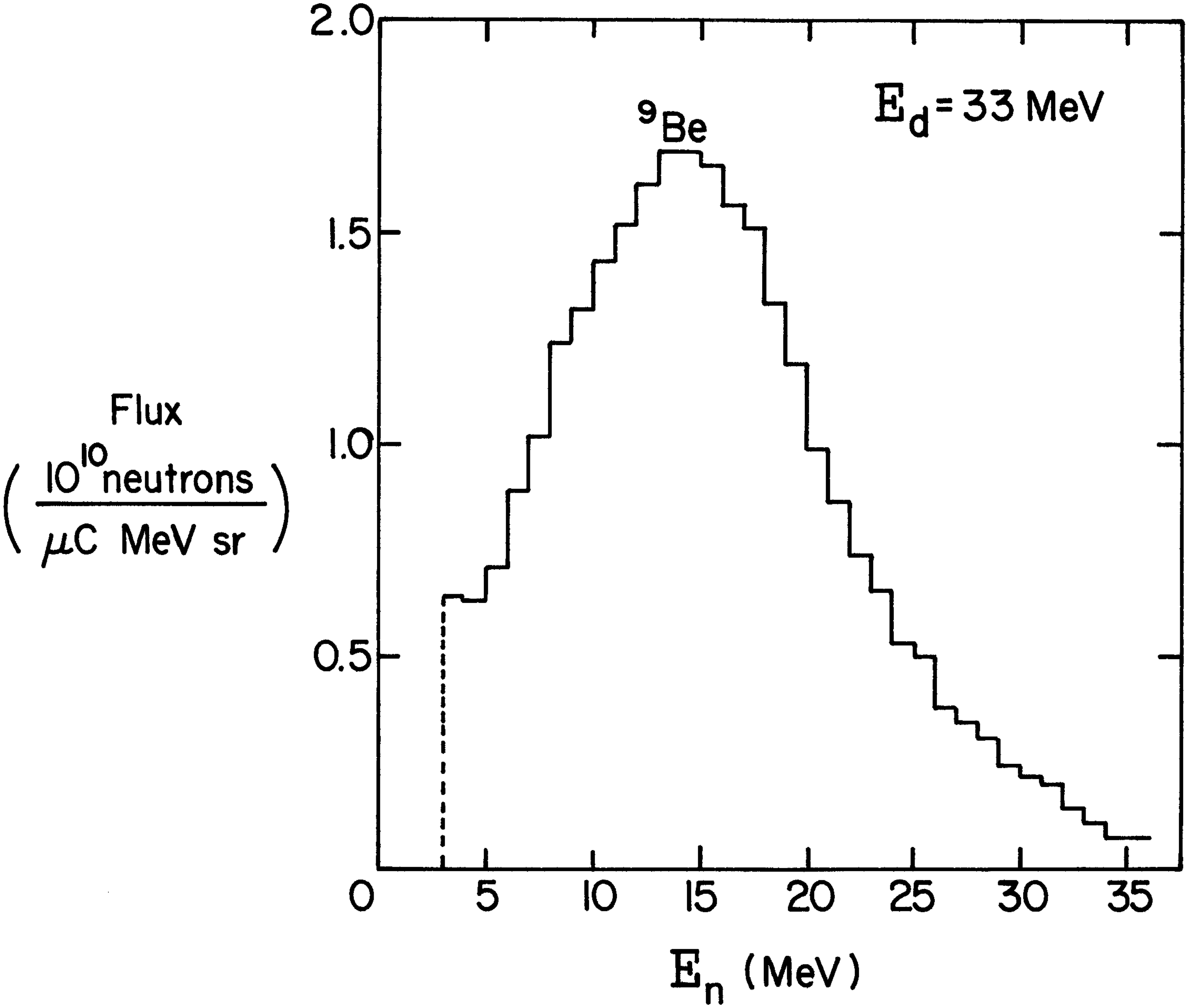


Figure 9

Spectrum of neutrons emitted in the forward direction in the bombardment of a thick beryllium target by 33 MeV deuterons. Only neutrons of energy greater than 4 MeV were observed (from reference 18).

Be target rotates at a speed such that the linear velocity at the beam position is 10 m/s, it is estimated that a temperature pulse of about 350°C would be produced.[19] A metallic Be target should be capable of withstanding such a temperature pulse.

A summary of the pertinent D + Be source parameters is also given in Table 14. We have assumed that the useful test volume is twice that of LINS because the flux assumed is at 1 cm from the source compared to 0.4 cm from LINS. At the larger distance, the inhomogeneity if the flux is less and therefore more (or larger) specimens can be tested.

V. Possible Scenario to Satisfy Needs Requirements

Now that we have established the fluence, dpa, gas production and test matrix information required for various reactors, let us examine how we might use existing or near term facilities to obtain the information required for one of the most critical D-T fusion devices, FERF-I. Again we will be forced to make many assumptions about the scheduling, operation of the sources and response of the materials. However, we feel that such an exercise is worthwhile at this time to focus on the perilous position we now face.

First of all what are we aiming at? Table 15 lists the information we would like to have by FY 1981 for both austenitic steel and aluminum. Table 16 describes how many years would be required to achieve the accumulated 14 MeV neutron fluences, dpa value, and helium content numbers for a LINS and a D + Be source. It can be seen that it takes only 2.5 years in a LINS to accumulate the 14 MeV neutron fluences in steel but 4.4 years to accumulate the required helium content and 6.4 years the required dpa rate. These numbers go from 2.5 for fluence to 6.1 for He and 13.1 for displacement damage in aluminum.

The numbers are more difficult to estimate for the D + Be source because of unknown cross sections above 15 MeV. However, if we assume constant dpa and gas production cross sections above 15 we come to the following conclusions.

The time required to accumulate the high energy ($E \geqq 14$ MeV) neutron fluence of $7 \times 10^{20} cm^{-2}$ is ~6-7 days. This number falls to 2 and 4 days to accumulate the required dpa damage in steel and aluminum respectively. Considering the helium production we find that between 6 and 10 days are required to obtain the required gas concentrations for steel and aluminum respectively.

It is now possible to estimate the number of machine years required for our hypothetical case. Using the information in Tables 6-12 we find that approximately 82 LINS neutron sources would be required (27 for SS and 55 for Al) and half that

TABLE 15

SUMMARY OF INFORMATION REQUIRED BY FY-1981 FOR AUSTENITIC STEEL AND ALUMINUM TO BE USED IN FERF-I

	Steel	Aluminum
Accumulated 14 MeV n Fluence - 1 year	7×10^{20}	7×10^{20}
dpa - 1 year	4.9	8.5
appm - He - 1 year	101	206
Temperature range	50-200	50-200
# of post irradiation tests	91	171
# of insitu tests	12	30
Volume of post irradiation tests$^{(a-d)}$ -cm^3	129	289
Volume of insitu tests -cm^3	36	102
Sub-total Volume -cm^3	165	391
Total -cm^3	556	
Number of post irradiation capsules	17	20
Number of insitu test capsules	12	30

a) assume tensile creep and fatigue specimens are 2 cm long, 1 cm wide and 0.1 cm thick
b) assume TEM specimens are 0.3 cm diameter and 0.01 cm thick, immersion specimens are 0.5 cm diameter and 0.5 cm high.
c) assume crack propagation and SN specimens are 0.5 x 0.5 x 2 cm.
d) assume individual temperature control capsules are 10 times volume of specimens.

TABLE 16

Summary of Assumptions Used in Calculating Machine Year Requirements for Accelerator Type Neutron Sources

LINS Data

Assume - 10^{13} n cm^{2} sec^{-1} (14 MeV)

- 90% duty factor
- 2.8×10^{20}n cm^{2} $year^{-1}$ (14 MeV)
- dpa rate per year, 0.76 for steel, 0.65 for Al
- He production rate appm $year^{-1}$, 23 for steel, 33.6 for Al
- 0.8 cm^{3} testing volume

	SS	Al
Number of years to accumulate 7×10^{20}n cm^{-2} (14 MeV)	2.5	2.5
Number of years to accumulate required dpa	6.4	13.1
Number of years to accumulate required He	4.4	6.13

D + Be Data

Assume - 3×10^{15} n $cm^{-2}s^{-1}$ E_n >1 MeV

2.3×10^{15}n $cm^{-2}s^{-1}$ E_n >10 MeV

1.4×10^{15}n $cm^{-2}s^{-1}$ E_n >15 MeV

- 90% duty factor
- dpa rate per year 120 for steel, 100 for Al
- He production rate, appm yr^{-1}, 450 steel, 650 Al
- 1.6 cm^{-3} testing volume

	SS	Al
Number of years to accumulate 7×10^{20}n cm^{-2} (E $\geqq$ 14 MeV)	0.02	0.02
Number of years to accumulate required dpa	0.004	0.008
Number of years to accumulate required He	0.02	0.03

Table 17

Approximate Number of Machine Years Required to Qualify Materials for the FERF-I Reactor

	Steel		Aluminum	
	LINS	D+Be	LINS	D+Be
Total Number of Machines - if parallel tests	27 (3 mixed)	14 (2 mixed)	55	28
Measure of Damage - Machine Years				
Accumulated 14 MeV neutrons	68	0.3	138	
Accumulated dpa	173	0.06	721	
Accumulated Helium	119	0.3	337	
Year Testing Must Start to Obtain Data by FY 1981				
Accumulated 14 MeV neutrons	FY-78 [a]	[b]	FY-78 [a]	[b]
Accumulated dpa	FY-74 [a]	[b]	FY-68 [a]	[b]
Accumulated Helium	FY-76 [a]	[b]	FY-75 [a]	[b]

(a) assuming all parallel tests

(b) tests would probably be done in series and therefore the date would depend on the number of machines built number of D + Be sources if all the tests were done at one time. However, a more realistic case might be to run tests in series in which case the number of machine years is important. This number depends on whether we use accumulated 14 MeV fluence, dpa or helium gas accumulation as a measure of damage. The machine years required for the various criteria are listed in Table 17 for steel and aluminum. A detailed list of the LINS devices and proposed functions is given in Appendix B.

If we were simply trying to duplicate dpa values, it would be necessary to start with 27 machines devoted to SS in FY-74 and 55 LINS machines devoted to Al in FY-68! If we were trying to duplicate helium production then we need to start with the same number of machines in FY-75 and 76. Finally, if it is only necessary to duplicate the uncollided 14 MeV fluence*, then we have until FY 1978 to construct the necessary 82 machines (all of this assumes no time for data analysis).

One could envision one or two or three irradiation testing facilities which would have 82, 41 or 28 LINS machines each. The facilities could have common management, tritium handling facilities, hot cells, maintenance crews, etc. The total costs of such facilities might be on the order of ~200 million dollars (~1/2 of the estimated cost of FERF-I or TFTR). Such a cluster of facilities might consume 20-30 MW_e for the power sources.

The critical time for initiation of D-Be tests does not depend on how many could be built because of the short time to accumulate the necessary data which means that the tests will probably be conducted in series, not in parallel. For example, if we assume that 2 accelerator sources could be devoted to such studies, then testing would only have to start in FY-80 to have data available by FY-81. If only 1 source were available then testing could start by FY-79. Of course,

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*The time difference between uncollided 14 MeV flux and He production mainly has to do with the use of back scattered neutrons in FERF to produce helium. There are assumed to be no back scattered neutrons in the accelerator sources.

one would want to build some associated D-T sources to compare selected data points such that designers could rely on the data obtained by this simulation technique. Given sources in FY-78, in addition to 2 d +Be sources, the total cost of such a facility would be approximately 50 million dollars.

Conclusions

Several points emerge from this rather complicated exercise in anticipating the needs of the near term D-T fusion reactors. The overall theme is that the current schedule to a Demonstration Power Reactor is quite ambitious and we may not be able to keep to that schedule if the near term EPR's and FERF's are delayed. More specifically:

1.) We are in danger of not being able to gather enough radiation damage information on the materials for the FERF-I so that a final design can be achieved by FY-81.

2.) Solid target D-T sources may be able to satisfy the requirements for assessing 14 MeV neutron damage only, but without a back scattered spectrum it is already too late to duplicate the required dpa and helium accumulation values by FY-81 for 1 year of FERF-I operation.

3.) Fast fission reactors could easily duplicate the displacement damage but can provide neither information on 14 MeV neutron damage nor duplicate the helium production. Furthermore the FFTF is hampered in that the lowest temperature attainable (315°C) is well above the anticipated operating temperature of FERF-I.

4.) Thermal reactors could be used to provide both the dpa and helium effects for steel but could not simulate the PKA spectra due to 14 MeV neutrons. Thermal reactors could not produce the proper helium rates in any other element. Furthermore, the current test reactors are not instrumented for the 40-50 of in situ tests which need to be performed on the structural material alone. Construction of new test reactors would take 5-10 years from the time of funding (another 2-3 years) and could not provide information by FY-81.

5.) D-Be sources may provide a partial answer to this dilemna in that, if one or two were constructed by the late 1970's, one could satisfy the need for radiation damage data in the FY-81-84 period. However, there still would be a need for D-T facilities to make sure that the data from the D-Be sources is typical of fusion neutron damage.

It is concluded that the most practical and economical solution to the problem is to construct a number (10-20) of solid target D-T sources and one or two D-Be sources by FY-78. These devices will accumulate the proper 14 MeV fluence, dpa, and required helium exposure by FY-81. This information coupled with thermal and fast fission reactor data could form the basis for final design decisions on FERF in FY-81. Continued irradiation with D-T sources into the 1980's could then confirm the materials behavior by FY-85 before actual fusion reactors will have operated.

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Figure Captions

Fig. 1 Summary of USAEC proposed plan to a tokamak demonstration power plant.

Fig. 2 Accumulated uncollided 14 MeV neutron fluence in various experimental tokamak fusion reactors. The asterisks give the time at which data at indicated fluence are needed for design of reactor.

Fig. 3 Accumulated displacement damage in steel in various experimental tokamak fusion reactors. The asterisks give the time at which data at indicated damage are needed for design of reactor.

Fig. 4 Accumulated helium (a) in steel and (b) in aluminum in various experimental tokamak fusion reactors. The asterisks give the time at which data at indicated helium concentration are needed.

Fig. 5 Deformation mechanism map for annealed 316 stainless steel.

Fig. 6 Simplified deformation mechanism map for annealed 316 SS. Dislocation creep and Coble creep are included in a thermally activated flow.

Fig. 7 Simplified deformation for aluminum alloy.

Fig. 8 Neutron flux expected from Livermore Intense Neutron Source for a neutron source strength of 4×10^{13} s^{-1} and 1 cm diameter deuteron beam. The flux is cylindrically symmetric around axis of abscissae. The contour marked 0.3 gives the distance within which the neutron flux exceeds $0.3 \times 4 \times 10^{13}$ cm^{-2} s^{-1}.

Fig. 9 Spectrum of neutrons emitted in the forward direction in the bombardment of a thick beryllium target by 33 MeV deuterons. Only neutrons of energy greater than 4 MeV were observed (from reference 18).

Appendix A

Abbreviations and Names for Tokamak Experiments and Proposed Reactors

Device	Name	Plasma Current (MA)	Laboratory and Country
PLT	Princeton Large Torus	1.4	Princeton Plasma Physics Lab (U.S.A.)
PDX	Poloidal Divertor Experiment	0.5	Princeton Plasma Physics Lab (U.S.A.)
TFTR	Tokamak Fusion Test Reactor	2.5	Princeton Plasma Physics Lab (U.S.A.)
D-III	Doublet-III	5.0	General Atomic Co. (U.S.A.)
EPR	Experimental Power Reactor		
FERF	Fusion Engineering Research Facility		
DPR	Demonstration Power Reactor		
CPR	Commercial Power Reactor		

Appendix B

An Example of the Number of LINS Units Which Would be Required to Adequately Test the Structural Materials (Steel and Aluminum) for FERF-I[a]

Machine #	Material	Temperature of Samples °C	Years of Operation When Samples Removed	Comment
1	SS & Al	50	0.6, 1.2, 1.8, 2.5	Swelling samples
2	"	125	" " " "	Swelling samples
3	"	200	" " " "	Swelling samples
			1.2, 2.5	2 Tensile samples
4	SS	50	1.2, 2.5	4 Tensile samples
5	"	"	"	"
6	"	"	"	"
7	"	"	"	"
8	SS & Al	"	"	Tensile (2 SS and 3 Al)
9	SS	200	"	4 Tensile samples
10	"	"	"	"
11	"	"	"	"
12	"	"	"	"
13	SS & Al	"	"	Tensile (2 SS and 3 Al)
14	SS	"	2.5[b]	SN (20 ksi)
15	"	"	"	SN (30 ksi)
16	"	"	"	SN (40 ksi)
17	"	"	"	SN (50 ksi)
18	"	"	"	2 Fatigue crack growth spec.
19	"	"	"	"
20	"	50	2.5[b]	Creep (5 ksi)
21	"	"	"	" (10 ksi)
22	"	"	"	" (15 ksi)
23	"	"	"	" (30 ksi)
24	"	200	"	" (5 ksi)
25	"	"	"	" (10 ksi)
26	"	"	"	" (15 ksi)
27	"	"	"	" (30 ksi)
28	Al	50	1.2, 2.5	4 Tensile samples
29	"	"	"	"
30	"	"	"	"
31	"	"	"	"
32	"	"	"	2 tensile samples
33	"	100	"	"
34	"	"	"	"
35	"	"	"	"
36	"	"	"	"
37	"	"	"	"
38	"	"	"	"

(a) Assume uncollided 14 MeV flux of 3.4×10^{20} n/cm^2 (at 90% Plant Factor)
(b) Insitu tests

Appendix B (cont.)

Machine #	Material	Temperature of Samples °C	Years of Operation	Comment
39	"	"	"	"
40	"	"	2	"
41	"	150	"	4 tensile samples
42	"	"	"	"
43	"	"	"	"
44	"	"	"	"
45	"	"	"	"
46	"	"	"	"
47	"	"	"	2 tensile samples
48	"	"	"	"
49	Al	200	1.2, 2.5	4 Tensile samples
50	"	"	"	"
51	"	"	"	"
52	"	"	"	"
53	"	"	"	"
54	"	"	"	"
55	"	"	"	"
56	"	"	"	2 Tensile samples
57	"	50	2.5[b]	SN (0.5 ksi)
58	"	"	"	SN (1 ksi)
59	"	"	"	SN (2 ksi)
60	"	"	"	SN (4 ski)
61	"	150	"	SN (0.5 ksi)
62	"	"	"	SN (1 ksi)
63	"	"	"	SN (2 ksi)
64	"	"	"	SN (4 ksi)
65	"	50	2.5[b]	Insitu crack prop.
66	"	200	"	"
67	"	50	"	Creep (1 ksi)
68	"	"	"	" (2 ksi)
69	"	"	"	" (4 ksi)
70	"	"	"	" (8 ksi)
71	" [	100	"	" (1 ksi)
72	"	"	"	" (2 ksi)
73	"	"	"	" (4 ksi)
74	"	"	"	" (8 ksi)
75	"	150	"	" (1 ksi)
76	"	"	"	" (2 ksi)
77	"	"	"	" (4 ksi)
78	"	"	"	" (8 ksi)
79	"	200	"	" (1 ksi)
80	"	"	"	" (2 ksi)
81	"	"	"	" (4 ksi)
82	"	"	"	" (8 ksi)