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MATERIALS PROGRAM PLAN FOR INERTIAL CONFINEMENT FUSION

by

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MASTER

FUSION POWER PROGRAM

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TABLE OF CONTENTS

	<u>Page</u>
1. Introduction	1
1.1 Alloy Selection.	1
1.2 Environmental Parameters	3
1.3 Program Plan Approach.	4
2. Microstructural Evolution.	10
2.1 Scope.	10
2.2 Background	10
2.2.1 Steady-State Irradiation	14
2.2.2 Pulsed Irradiation	15
2.3 Data Requirements and Strategy	20
2.3.1 Materials Test Facility.	20
2.3.2 Experimental Program	24
3. Fatigue and Mechanical Properties.	32
3.1 Scope.	32
3.2 Problem Definition and Analysis.	33
3.2.1 Mechanical and Thermal Environments.	33
3.2.2 Chemical and Irradiation Environments.	36
3.3 Data Requirements.	36
3.3.1 Fatigue.	36
3.3.2 Other Mechanical Properties.	41
3.4 Strategy	42
4. Lithium Compatability and Other Environmental Effects.	57
4.1 Scope.	57
4.2 Background	58
4.3 Data Requirements.	60
4.3.1 Lithium Compatibility.	60
4.3.2 Other Environmental Effects.	64
4.4 Strategy	64
5. Executive Summary.	70

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1. INTRODUCTION

A materials development plan is a key ingredient in the definition of a pathway to a commercial inertial fusion power plant. The objective of this report is to lay out a general strategy that will support the construction of both nearer term devices such as the Engineering Test Facility (ETF), and the Experimental Power Reactor (EPR) as well as the first commercial plant. The schedule assumed for these devices is that presented in the Deutch¹ report of September 1978 and is reproduced in Fig. 5-1 of the Executive Summary section.

Other major assumptions used in this planning effort include:

1. A continuing LMFBR R&D program.
2. An expanding magnetic fusion² materials program, based on the recent task force planning effort.
3. A continuing Basic Energy Sciences program that will explore fundamental questions on radiation effects, lithium compatibility and mechanical properties.
4. The presence of some scheme to protect the first structural wall from short range radiation and particles with deposition times shorter than a microsecond.

The present report concentrates on the structural requirements of the reactor vessel and is done in a manner to make it as independent of driver choice as possible. Every attempt was made to find a cost-effective developmental path that concentrated on obtaining sufficient information to allow timely engineering decisions. Planning this path for an emerging technology, such as inertial confinement, must be a continually dynamic process.

Two key strategy questions were posed in the planning for each technical area. These were:

1. Is a major separate materials test facility required by this program? If so, when and at what approximate cost?
2. What are the specific areas of the required materials technology development that the inertial confinement program must consider won't be sufficiently supported by other energy technologies?

1.1 Alloy Selection

The EPR is expected to be the first large volume, high plant factor test facility available for generating the statistical data on materials response

required for engineering design. As such, it is felt to be unlikely that any alloy or alloy system that is not now already in the process of being well characterized will be used in the construction of either ETF or EPR. The approach taken in this development plan, therefore, is to assume the use of alloy systems for which a commercial industry currently exists, for which joining and fabrication technology is known, and for which there is an ongoing program to establish their response to a fast neutron environment.

Based on this premise, the suggested materials development plan concentrates on two classes of steels for application to ETF and EPR. More advanced systems, based on such elements as vanadium, niobium or titanium, would be developed through the use of the latter device as an irradiation test bed. The two classes of steels recommended to receive the most near term attention are the ferritic stainless steels as the leading candidate, and the austenitic steels as a back-up. The ferritic stainless steels are felt to be particularly appropriate for those fusion reactor designs that use some form of liquid lithium first wall protection. They may not be appropriate for designs that use magnetic deflection for protection.

Within the broad classification of ferritic stainless steels, two major categories are suggested. The first is based on a composition range of from 9-12%Cr, 1-2%Mo with small additions of other elements such as V, Nb and W. These steels, which are heat treated to form a tempered martensite micro-structure, are expected to be stable in a lithium environment, are resistant to radiation-induced void formation, show promise of being little affected by helium embrittlement, have strengths better or comparable to the austenitic stainless steels up to temperatures of $\sim 600^{\circ}\text{C}$ and develop about half the thermal stress for a given surface heat flux than do the austenitic steels.

Additionally, they have a significantly lower biological hazard potential than the austenitic steels after decay times of 10 years or more. Commercial production of alloys in this class has centered on their application to superheater tubing in fossil power plants in Europe. Sandvik fabricates a 12%Cr alloy designated HT-9 for this application while Vallourec has developed a 9%Cr steel, Em-12, for a similar purpose. Since 1974 the U. S. LFMBR program has sponsored a joint developmental effort at Combustion

Engineering and Oak Ridge National Laboratory³ to improve the toughness, thermal stability and weldability of these alloys that has resulted in major property improvements. This class of steels has received much recent interest for application in an irradiation environment by both the LFMFR and magnetic fusion programs.

The second type of ferritic alloy is based on the 2-1/4%Cr-1%Mo compositional base. This steel shares many of the advantages of the 9%Cr steels with respect to radiation damage resistance, lithium compatibility, and levels of thermal stresses generated under a given heat loading. It has lower creep strengths than some of the 9%Cr steels and is susceptible to rusting, which is a disadvantage during field erection. The 2-1/4%Cr-1%Mo steels have the advantage of a strong domestic commercial experience base plus the results of an extensive research and development program supporting their selection as the reference LMFBR steam generator material.

The austenitic stainless steels as typified by Type 316 are included as back-up candidates in the event that radiation effects on the ductile-brittle transition temperature of ferritic steels result in unacceptable toughness levels at temperatures of interest. Type 316 is the most highly researched structural alloy for irradiation use, is widely available and easily fabricated and, thus, represents a fall-back position for the program.

1.2 Environmental Parameters

The expected flux levels, energy spectra and duration times of neutron, ion, and electromagnetic radiation that may be incident on the first surface are given in Table 1-1. The spread in these values is a result of both the assumption of different possible reactor design concepts (e.g. lithium waterfall vs. wetted wall), and the pellet design and chamber gas pressure dictated by the use of different drivers. The establishment of the assumed ranges was based on work at Livermore,⁴ Exxon,⁵ Los Alamos,⁶ Sandia,⁷ and Wisconsin,⁸ as well as on calculations conducted separately at ANL.^{9,10} The interrelationship between design approaches and materials requirements is particularly important for inertial confinement development. After considering materials development requirements, one of our conclusions is that the materials effort must be closely coupled with the reactor design development effort. This appears to be an essential ingredient in finding a

Table 1-1. Environmental Parameters

	Peak Flux (cm ⁻² s ⁻¹)	Energy Spectrum	Deposition Time
Neutrons	10 ²⁰ -10 ²¹	14 MeV G [*] <E> = 12.6 MeV [†]	1-50 ns
Ions	10 ¹⁸ -10 ¹⁹	300-800 keV M	1-10 μs
X-rays		0.3-1 keV BB	10-100 ps

* Spectral type

BB = Blackbody

M = Maxwellian

G = Gaussian

† <E> is the average energy

cost-effective approach to a commercial power system based on the challenging conditions presented by inertial confinement fusion. A summary, based primarily on the work of Maniscalco⁵, of many of the relevant materials requirements resulting from the different designs is presented in Table 1-2. In this materials development planning study, some form of first wall surface protection is assumed so that the flux of ions to the first structural surface is minimized, as is the power deposited in the front surface by electromagnetic radiation.

1.3 Program Plan Approach

The program plan is laid out in five sections. Following the first introductory section, the second section deals with the effect of the irradiation environment on the microstructure of materials. A major part of the initial activity in this area will be aimed toward evaluating the importance of pulse effects on microstructural development. The usefulness of existing test facilities in determining the importance of pulse effects will relate directly to the need and desirability of building a dedicated materials test facility. The approach taken in this and subsequent sections is to define the expected situation in general terms, identify the data required, point out what information is expected from other sources, and finally, present a schedule and discuss the manpower and facilities that the

Table 1-2.

ICF Reactor Design Parameters Which
Influence Material Requirements

I. Driver and Target Parameters	Dry Wall	Wetted Wall	Magnetic Deflection	SOLASE	HYLIFE	Light Particle Beam
Target Gain	100	130	130	150	900	30
Target Energy Yield, MJ	100	130	130	150	2700	60
Pulse Repetition Rate, s^{-1}	7	1	10-20	20	1	10
Driver Energy, MJ	1	1	1	1	3	2
Driver Efficiency, %	3	5-10	5-10	6.7	3	20-30
II. First Wall and Cavity Design						
Geometry	Cylinder	Sphere	Cylinder	Sphere	Cylinder	Cylinder
Radius, m	5	1.5-3	2-5	6	5	2
Height, m	10	Na	4-10	Na	8	4-5
Background Cavity Pressure, MPa	10^{-5}	$10^{-5} - 10^{-4}$	$10^{-6} - 10^{-5}$	6.7×10^{-5}	$<10^{-5}$	6.7×10^{-3}
Peak Cavity Pressure, MPa	10^{-4}	2×10^{-1}	10^{-1}	1.2×10^{-2}	---	2×10^{-2}
First-Wall Neutron Flux, MW/m^2	1.6	1-3.5	3-15	5	6.9 midplane	8
Coolant	liquid Li	liquid Li	liquid Li	Li_2O	liquid Li	Li
Front Surface Materials	graphite	liquid Li	graphite	graphite	liquid Li	ferritic steel
Structural Materials	Mo	Nb or SS	Nb or SS	graphite	ferritic steel	ferritic steel
Na - not applicable						

Table 1-2. (Continued)

First Wall and Cavity Design (Continued)	Dry Wall	Wetted Wall	Magnetic Deflection	SOLASE	HYLIFE	Light Particle Beam
Desired Lifetime of First-Wall, yr.	1	3	1	1	30	2-3
Maximum Stress in Front Surface, MPa	80	Na	---	41.4	---	75
Maximum Stress in First- Wall Structure, MPa	30	100-200	100-200	19.7	89	60
Maximum Front Surface Temperature, °K	1800	Na	700-1270	2270	Na	900
Average Front Surface Temperature, °K	1200	570-770	570-770	2120	770	750
Maximum Structure Temperature, °K	750	570-770	570-770	1020	800	750
Displacement Damage in Front Surface, dpa/yr.*	16	Na	---	32.5	Na	---
Displacement Damage in Structure, dpa/yr.*	11	10-30	40-200	32.5	2.9 midplane	---
Helium Production in Front Surface, at ppm/yr.*	3000	Na	4200-21,000	7000	Na	---
Helium Production in Structure, at ppm/yr.*	50	40-300	150-800	7000	3.4	---
III. <u>Blanket Design</u>						
Coolant	liquid Li	liquid Li	liquid Li	Li ₂ O	liquid Li	liquid Li

Table 1-2. (Continued)

Blanket Design (Continued)	Dry Wall	Wetted Wall	Magnetic Deflection	SOLASE	HYLIFE	Light Particle Beam
Pressure (atm)	---	3	3	---	---	---
Structural Materials	ferritic steel	SS or Mo	SS or Mo	graphite, Al	ferritic steel	ferritic steel
Desired Lifetime of Structure, yr.	5	30	30	1	30	30
Maximum Stress in Structure, MPa	30	100-200	100-200	32.4	30	60
Maximum Coolant Tem- perature °K	770	570-1070	570-1070	870	770	700
Maximum Structure Temperature °K	790	770-1070	770-1070	1020	800	750
Displacement Damage in Structure, dpa/yr.*	2-11	1-3	4-20	0.2-32.5	0.1-3.6	---
Helium Production, at ppm/yr.*	5-50	4-3000	15-800	18-7000	0.1-4.3	---

*to convert to SI units of events/s from events/yr divide by $3.156 \cdot 10^7$.

inertial fusion program must be willing to dedicate to provide the necessary information on a timely basis.

Section 3 deals with the development effort that is necessary to cope with the high cycle loading of the first wall structure. The loading pulses are expected to range from 1 to 20 per second (3×10^7 to 6×10^8 /year), thus creating a high cycle fatigue problem for any long-lived first wall structure. The interrelationship between specimen and component testing is a major issue in this section. Static mechanical property requirements are also considered here. Of particular importance are the combined effects of point defect clusters and the transmutation produced helium and hydrogen on the fracture toughness and ductility of candidate structural materials.

Lithium compatibility is treated in Section 4. It is likely that inertial confinement designs will be more dependent on the use of liquid lithium than magnetic fusion devices. This is due, on the one hand, to the fact that magnetic fields are no longer needed, thereby removing MHD effects on pumping and, on the other, on the need to provide a protective first wall layer against the extremely high X-ray and ion power levels that result from the short time scale of the energy release process. The possible effect of pellet debris on compatibility is a subject of special consideration due to its uniqueness to inertial confinement fusion. The facility requirements for this key area must be defined early in the program.

The final section integrates the conclusions reached in the body of the report into a unified strategy that suggests a particular effort level to support major program milestones.

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2. MICROSTRUCTURAL EVOLUTION

2.1 Scope

Mechanical properties of materials are strongly dependent on details of microstructure. In an inertial confinement fusion reactor (ICFR), the first wall will be subjected to displacement damage, thermal pulses, and stress pulses, all three of which can produce changes in microstructure. The purpose of this section is to discuss the features of the ICFR environment that lead to microstructural evolution in the first wall, indicate the materials problems that may arise, and outline the types of programs and facilities that are needed to assess and solve these problems. The ICFR environment is discussed first, along with the implications of various reactor designs on the approach taken in this study, and available experimental and theoretical data on the effects of the important components of the environment is reviewed. Then the data necessary to meet the ICFR programmatic goals is outlined, and a strategy for attaining the necessary information is suggested. Finally specific recommendations are made for materials testing programs based both on available testing facilities and on a possible, prospective dedicated test facility necessitated by large pulsed neutron effects or currently unforeseen testing needs.

2.2 Background

The materials environment of an ICFR is harsh. In an unprotected first wall, ions from the pellet debris can produce displacement damage at a rate as high as 100 displacements per atom (dpa) per second in the near surface region. X-rays can raise surface temperatures above the melting point, with thermal stress waves generated by the rapid energy deposition; the blast wave from the pellet implosion can cause pressure pulses that are capable of plastically deforming the first wall material. The instantaneous neutron dose rate following the pellet burn is orders of magnitude larger than that found in fission reactors. (See Table 2-1 for a comparison of the expected ICFR neutron environment with that in other facilities.) To compound the problem, the thermal and mechanical loadings and displacement damage are applied intermittently, providing an unique environment about which not much is currently known.

Table 2-1 Test Facilities

	ICFR ¹ (Li protected) (14 MeV neutrons)	RTNS-II ² <E> ~ 14 MeV neutrons	ICFR ³ (non Li protected) (14 MeV neutrons)	FMI ⁴ <E> ~ 14 MeV neutrons
Peak Displacement Rate, dpa-s ⁻¹	0.1	3×10^{-8}	1-10 (neutron only)	2×10^{-6}
Average Displacement Rate, dpa-s ⁻¹	10^{-7} - 10^{-6}	3×10^{-8}	10^{-6} - 2×10^{-5}	2×10^{-6}
Peak Particle Flux, cm ⁻² s ⁻¹	$\sim 5 \times 10^{20}$	10^{13}	$\sim 5 \times 10^{20}$	10^{15}
Displacement Pulse Width, ns	10	continuous	10	continuous
Pulse Repetition Rate, HZ	1	dc	10-20	dc
Helium Production in Structure, appm-yr ⁻¹	6*-300**	30	50^+ -7000 ⁺⁺	3000

¹HYLIFE, wetted wall concepts; ~ 100 MJ explosions for all ICFR's.

²To be available in 1979 for dedicated materials testing.

³Dry metal wall, magnetic deflection, and SOLASE designs.

⁴To be available in 1982.

*HYLIFE design.

**LASL wetted wall design.

[†]Dry metal wall.

⁺⁺SOLASE graphite wall.

Table 2-1. (Continued)

	Heavy Ion Accelerator (4 MeV nickel)	WNR ⁵ (spallation neutron source) (10 MeV avg.)	FFTF Fast Neutrons <E> ~ 0.45 MeV	IPNS-I ⁶ (fast neutron) E > 0.1 MeV <E> ~ 0.7 MeV	HVEM (1 MeV electrons)
Peak Displacement Rate, dpa-s ⁻¹	0.1-0.4 ^{†*}	0.05-0.10	10 ⁻⁷ -10 ⁻⁶	10 ⁻⁴ -10 ⁻³	10 ⁻³
Average Displacement Rate, dpa-s ⁻¹	10 ⁻⁷ -10 ⁻³	10 ⁻⁷ -10 ⁻⁶	10 ⁻⁷ -10 ⁻⁶	10 ⁻⁸ -10 ⁻⁷	10 ⁻³
Peak Particle Flux, cm ⁻² -s ⁻¹	10 ¹² -10 ¹³	2-5 x 10 ²⁰	10 ¹⁵	0.3 x 10 ¹⁸	10 ¹⁵
Displacement Pulse Width, ns	2	200	continuous	10,000	continuous
Pulse Repetition Rate, HZ	dc to 10 ⁷	1-10	dc	15-45	dc
Helium Production in Structure, appm-yr ⁻¹	available with dual beam in- jection	approx. same as ICFR	3-30	0.3-3	none

⁵WNR facility to be operating with proton storage ring and able to perform as shown by 1984.

⁶IPNS-I to be operational by 1982.

^{†*}0.1 dpa/s corresponds to 1 μ A beam focused to a 2 mm diameter beam of 25 μ A/cm² and 0.4 dpa/s to a 1 mm diameter beam with density of 100 μ A/cm².

An initial assessment of environmental parameters and realization of the severity of the bare wall environment has led to design concepts that include some protection for the first wall. A lithium-wetted wall, lithium waterfall, magnetic deflection, or a buffer gas prevents both X-ray and ion interactions with the wall and therefore eliminates a major portion of the surface damage. Thermal stress waves are ameliorated by the protection devices and are in the elastic regime,¹ thereby generating no massive changes in the microstructure. With the exception of the lithium waterfall proposed in the HYLIFE design, the neutron flux is essentially unaffected by the protection schemes; bulk irradiation damage that is due to the neutrons thus becomes the major concern and microstructural evolution will be primarily affected by the radiation component of the environment.

Neutron Effects

The ICFR neutron environment has significant differences from that of a fission reactor. The two major differences are its cyclic nature and the high instantaneous damage rates. Fission reactors operate in a steady-state mode with continuous neutron fluxes and damage rates of the order of 10^{-7} - 10^{-6} dpa-s⁻¹. The ICFR will cycle at 1-10 Hz with a neutron pulse lasting between 1 and 100 ns, producing displacements at rates between 10^{-2} and 10 dpa-s⁻¹. The lower damage rates are appropriate for waterfall designs, the higher rates for other protection schemes.

The interaction of neutrons with a material leads to several types of change in the material. Coupling to the electronic structure results in temperature changes, elastic collisions give displacement damage through cascade formation, and nuclear reactions produce helium and solid transmutants. The excess point defects (vacancies and interstitials) generated by displacement events can aggregate to form dislocation loops or voids, enhance precipitation or dissolution reactions, and cause redistribution of solute elements by enhanced diffusion or segregation. Solid transmutants can affect phase stability and harden the matrix. The products of the neutron-matrix interactions can thus cause nucleation of new microstructural features and enhanced growth or redistribution of those present before irradiation.

On the macroscopic scale, these types of microstructural changes can cause irradiation-induced deformation, i.e. swelling and creep, and loss of

ductility. The presence of helium can lead to bubble swelling at high temperatures and grain boundary embrittlement. If segregation of solute elements to surfaces occurs, the change in surface composition may affect corrosion properties. The severity of these potential materials problems in the ICFR can be assessed if microstructural changes in the pulsed neutron environment are understood.

2.2.1 Steady-State Irradiation

Studies of radiation damage that were initiated to support fission reactor technology can provide some of the information necessary for understanding radiation effects on microstructure and mechanical properties. The Liquid Metal Fast Breeder Reactor (LMFBR) program has produced a large quantity of data on steady-state neutron irradiation effects in materials at constant temperature. Empirical equations that describe steady-state swelling and creep in the austenitic stainless steels have been formulated;² ductility and stress-rupture data have been obtained from post-irradiation testing;³ and some information on precipitation effects⁴ and phase stability⁵ has been obtained. The program has focused primarily on austenitic stainless steels and the void-swelling phenomenon, with the aim of predicting its magnitude at high doses and tailoring materials to minimize its occurrence. Because of an emphasis on design requirements and the availability of a prototypic environment in EBR-II and other fast reactors, the general approach to radiation damage in reactor structural materials has been to subject the existing data on macroscopic dimensional changes to statistical analysis and obtain empirical correlations between, for example, swelling, dose, dose rate, stress, and temperature. This approach does not require a knowledge of the microstructural changes that neutrons induce, and consequently has not produced a large data base of detailed microstructural observations that are correlated with mechanical properties data.

Substantial understanding of microstructural changes caused by displacement damage has been obtained from theoretical models of individual phenomenon. The models have been used to identify important physical processes involved in microstructural evolution and to predict qualitative dependencies on irradiation parameters. The models, however, are not based on first principles and must be calibrated with detailed microstructural data from neutron irradiated materials in order to yield quantitative predictions.

These data are not generally available. Ion and electron beam irradiations,⁶ which can be used to simulate neutron irradiation, have provided data necessary to verify mechanisms and have been used to study, under relatively controlled conditions, the kinetics of microstructural changes.⁷ However no correlation of microstructural changes under different irradiation conditions is available to link the simulation and neutron results or to extend the current data base to the ICFR environment.⁸ Because ion and electron irradiations damage only the near surface region of a material, these techniques have limited use in the study of damage effects on bulk mechanical properties or of correlations between microstructures and mechanical properties.

It should be emphasized that the data available on neutron irradiation effects have been obtained under steady-state conditions, generally with flux, temperature, and stress all held constant, and typical dose rates are orders of magnitude lower than the instantaneous rates expected in an ICFR (10^{-7} dpa-s⁻¹ in EBR-II compared with 10^{-1} dpa-s⁻¹ in the HYLIFE design or 10 dpa-s⁻¹ in the SOLASE design). Some experiments involving single-step changes in temperature or stress with constant flux are in progress,⁹ but no systematic studies have been reported. Considerations of the effects of pulsed conditions on microstructural development have been very limited, with most of the work being theoretical using adapted steady-state computer codes. A few experiments with pulsed ion beams have been reported. This work and the results will be considered in detail later in the section.

2.2.2 Pulsed Irradiation

Planned Magnetic Fusion Effort

Many features of the ICFR first wall environment are also present in the magnetic fusion reactor (MFR) first-wall environment. The neutron energy spectra are similar, so generation rates for helium and solid transmutants will be comparable. Proposed operating temperature ranges are the same for both (300-500°C for steels and 500-1000°C for refractories). The anticipated stress levels in ICFRs are approximately a factor of two higher than those in MFRs, due to higher instantaneous heat loadings. MFRs are also operated in a pulsed mode, with $10^3 - 10^5$ power cycles per year for tokamak designs. In many areas, then, the materials concerns for MFR first walls must parallel those for ICFRs.

The Task Group on Damage Analysis and Fundamental Studies (DAFS), which provides technical assistance to the Materials and Radiation Effects Branch of the Office of Fusion Energy, has prepared a program plan for magnetic fusion materials development.¹⁰ In the area of damage microstructure evolution, the program plan emphasizes studies aimed toward understanding the interaction of helium with displacement damage, because of its anticipated effects on ductility, and development of methods to correlate damage microstructures with mechanical properties. The latter effort is necessary because of the small test volumes in available high-energy neutron facilities, and the fact that charged particle irradiations are best suited for microstructure studies. Assessments of the effects of hydrogen and solid transmutants on microstructure are also planned, along with the effects of cycling and damage rate. Most of the microstructural studies are scheduled to be completed between 1984 and 1986, and the validation and calibration of microstructural models of flow and fracture behavior during irradiation are to be finished between 1988 and 1990.

The scope of the magnetic fusion materials program is very broad, including development of techniques to characterize various irradiation environments, studies of the primary damage state, measurement of high-neutron energy cross sections for nuclear reactions and other parameters whose values are necessary for model development. Much of the information obtained from this program will benefit the ICFR materials effort.

There are two major differences between the radiation environments in MFRs and ICFRs, however, and these differences pose problems that are specific to inertial confinement devices. Current design concepts for ICFRs require power cycling at the rate of 10^7 and 10^9 cycles per year, which is several orders of magnitude larger than the rate expected in tokamaks. The duration of the damage pulse is ~ 10 ns, substantially smaller than the times anticipated for tokamaks, and the pulse amplitude is much larger. Instantaneous damage rates for tokamaks are between 10^{-7} and 10^{-5} dpa-s⁻¹; for ICFRs between 10^{-2} and 10 dpa-s⁻¹, a very different regime. At the higher dose rates cascade interaction or overlap may change the efficiency of point-defect production resulting in, for example, a smaller proportion of defects surviving the initial damage event and therefore fewer available to participate in microstructure changes. The impact of the duration difference can be

partially assessed by considering the limited amount of theoretical and experimental work that has been done on pulsed damage effects.

Recent Theoretical Efforts

The work on pulsed irradiation effects has been confined to model calculations of point-defect concentrations, void nucleation rates, and void growth rates, and experimental studies of void microstructures. Processes such as creep and enhanced diffusion are expected to be proportional to the time-averaged defect concentration, and aggregation or clustering reactions proportional to the time-averaged square of the defect concentrations. An analysis of the variation of vacancy and interstitial concentrations during a single pulse can thus be used to make qualitative comparisons with steady-state behavior. Dienes¹¹ calculated vacancy and interstitial concentrations as functions of time (or equivalently number of pulses) assuming either exponential decay to sinks or bi-molecular recombination or both as annealing modes. He found that the defect concentrations reached steady-state after a large number of cycles and compared the steady-state values with those obtained under continuous irradiation at the time-averaged damage rate. Defect concentrations averaged over the pulse cycle were generally the same or lower than the continuous irradiation concentrations, although the maximum was higher than the continuous irradiation result. The square of the defect concentration averaged over the pulse cycle was larger than that for continuous irradiation when only one annealing mode was considered or when the species was faster if both annealing modes were included. One infers, therefore, that processes whose rates depend on the average defect concentration will be slower during pulsed irradiation and those whose rates depend on the time-averaged square of the concentration may be faster or slower depending on the relative mobility of the species. These conclusions are general, since they are based only on the forms of the point defect equations and therefore apply to all pulsed irradiations.

Calculations done for this study¹² using parameters appropriate for 316 SS and ICFR and MFR pulse times show significant differences in the behavior of the point defect concentrations under the two sets of conditions. During the ICFR pulse cycle, the interstitial concentration rises linearly, then anneals completely; no ratcheting occurs as the number of cycles increases. The vacancy concentration rises until the end of the irradiation, drops

slightly at the beginning of the annealing phase due to recombination with interstitials, then remains essentially constant until the beginning of the next pulse cycle. The annealing period is short compared with the vacancy lifetime, and hence ratcheting of the vacancy concentration is rapid. Under MFR conditions, on the other hand, irradiation time is long enough so that the vacancy and interstitial concentrations approach their steady-state values before the annealing phase begins, during which the concentrations of both defects drop. The differences between ICFR and MFR conditions may be more clearly illustrated if a specific phenomenon such as interstitial loop nucleation is considered. Rate theory calculations of interstitial loop number densities indicate that nucleation is nearly complete at a dose of $\sim 10^{-4}$ dpa for a wide range of temperatures and dose rates. For a time of 10^{-8} s and a dose rate of 10 dpa-s^{-1} , it is clear that under ICFR conditions this dose is not reached during a single pulse; for MFR parameters, however, the dose during a single pulse is 10^{-4} or larger. The average interstitial supersaturation remains low in an ICFR compared with that in an MFR. Thus an interstitial loop microstructure may develop and grow early in an MFR environment and be retarded in an ICFR. Arguments of this sort provide some insight into microstructure evolution, but many important defect reactions, such as impurity trapping, clustering, radiation-enhanced reaction rates, and segregation, depend on vacancy and interstitial concentrations in a complex way and can only be investigated by detailed computer calculations.

Odette and Myers¹³ have calculated void nucleation rates under pulsed conditions and compared the results with rates obtained for continuous irradiation at the instantaneous pulse damage rate. They found, as might be expected, that nucleation rates decreased as pulse times decreased and annealing times increased. In their calculations, however, they assumed that both vacancy and interstitial concentrations varied in phase with the irradiation pulse, which is equivalent to assuming that both concentrations reach and decay from steady-state values in a time that is short compared with the irradiation and annealing times. For MFE conditions this is a reasonable assumption, especially for the interstitial concentration; for the ICFR conditions, however, this assumption does not hold, and extension of their conclusions to the ICFR pulse regime must be justified.

The nucleation and growth of interstitial loops and vacancy clusters under ICFR conditions is being calculated with a rate theory model that

considers both defect structures simultaneously.¹² Preliminary results for several pulse cycles show that loop development is retarded when compared with that obtained under continuous irradiation at the time-average damage rate.

Several authors have applied void swelling equations, which were developed to explain steady-state irradiation results, to pulsed irradiation conditions. Ghoniem and Kulcinski¹⁴ analyzed the effect of a single pulse on the void radius. They found that void size generally decreased, except when pulsing was timed so that irradiation resumed after all vacancies produced by the previous pulse were annealed, but before voids began annealing due to thermal emission. Kmetyk, et al.,¹⁵ extended Ghoniem and Kulcinski's work to include multiple pulses and temperature oscillations. Their calculations for Al irradiated with a pulse of 0.5 ms at a frequency of 120 Hz showed a constant void growth rate after the vacancy and interstitial concentrations reached an oscillatory steady-state. The growth rate decreased significantly from that for continuous irradiation at the time-average damage rate only when a temperature excursion, due to neutron heating, was included in the dynamic equations. This work, then, as well as the other work on void nucleation and growth under pulsed irradiations suggests that void swelling may be less severe in an ICFR than in, for example, a fission reactor with the same average dpa rate. However, calculations have not been done for potential first-wall materials or for the specific conditions proposed in ICFR designs. No calculations have been done to analyze the behavior of phases, precipitates, or the dislocation structure under ICFR conditions.

Experimental Results

A limited amount of experimental data, obtained from ion irradiations, is available for comparison with model calculations. Taylor, et al.,¹⁶ have investigated the effects of pulsed Ni^+ -ion irradiations on the void microstructure in pure Ni. The void microstructure developed with a pulsed beam had a lower number density and a larger average void size than that which developed with a continuous beam of the time-average intensity, suggesting that pulsing reduces the void nucleation rate and enhances the void growth rate. A series of experiments with pre-voided samples was performed to determine the effect of beam structure, that is, relative on-off times, on void growth. A long on-short off beam structure produced the same swelling as a constant beam; very little swelling occurred for a short on-long off structure. The greatest increase in swelling occurred for equal on-off times.

Kaletta¹⁷ has done transmission electron microscopy (TEM) analysis of the bubble population in vanadium implanted with helium in both continuous and pulsed modes. For pulse times between 300 and 600 s, he found that bubble swelling at low doses (<1 dpa) was increased by nearly an order of magnitude under pulsed conditions. He postulates that relaxation of small helium clusters during the annealing period increases the concentration of small helium bubbles and leads to increased bubble coalescence in the next irradiation period. This result points to a potential problem for ICFR first walls; however, the time structure of the pulses and the irradiation conditions for these experiments corresponded to irradiation pulses long compared with the interstitial lifetime but shorter than the vacancy lifetime. Since ICFR pulse times are on the order of the interstitial lifetime, these results do not necessarily apply to ICFR conditions.

The summary given above supports two conclusions: First, different regimes of microstructural behavior may be defined by the different pulse times and duty cycles in MFRs and ICFRs, and second, microstructural modeling and ion bombardment are two major tools readily available for pulsed irradiation studies. They are, in fact, currently being used to assess potential problem areas.

2.3 Data Requirements and Strategy

In this section, the two questions posed in the Introduction are addressed, and a program for investigating the effects of pulsed irradiation investigating on microstructural evolution in the ICFR environment is outlined.

2.3.1 Materials Test Facility

The need for a dedicated Materials Test Facility (MTF) depends on the relative magnitude of pulsed effects on microstructural evolution. If pulsed neutrons cause only small and understandable changes when compared with those obtained during steady-state irradiation, then the EPR can be confidently built using data obtained from steady state irradiation devices (accelerators and fission reactors). If, however, the pulsed nature of the radiation damage causes major and inexplicable changes in microstructure, a pulsed materials test facility may have to be constructed at the same time as ETF in order to supply the data to build an EPR.

From the summary given in Section 2.2.2 it is clear that the effects of pulsed irradiation on microstructure are not well understood. More information must be obtained, especially on effects in the ferritic and austenitic stainless steels proposed for the near-term devices, before a decision on the necessity of an MTF can be made. The suggested program that allows the scoping of this problem by 1987 relies on the combined use of currently available experimental facilities and an associated theoretical modeling effort.

As can be noted in Table 2-1, several possible test beds are available.

Existing radiation test facilities include the following categories: heavy ion accelerators,¹⁸ high flux neutron sources such as the WNR facility at Los Alamos,¹⁹ or IPNS-I at Argonne,²⁰ fast reactors such as FFTF,²¹ RTNS-II at Livermore,²² the Fusion Materials Irradiation Test Facility (FMIT),²³ and high energy electron sources in the form of high voltage electron microscopes (HVEM).²⁴ Some of these only produce microstructural damage near free surfaces (heavy ion accelerators and HVEM) and some bulk damage (WNR and FFTF). In Table 2-1 ICFR irradiation conditions are compared with a variety of radiation test facilities available now or in the near future.

The test facilities listed may be useful if several conditions are satisfied: (1) the simulation scheme devised must either closely approximate expected ICFR radiation conditions or provide a damage structure which can be related by standards or modeling to an ICFR damage structure; (2) the test facilities must be readily available a large fraction of the time since a complete materials test program could extend over many years and could involve large numbers of test specimens which require well defined experimental conditions; and (3) the economics of the simulation method should not be prohibitive.

The heavy ion accelerator is capable of providing both damage rates (up to about 0.4 dpa-s^{-1}) and pulsed conditions (pulse widths as low as 50 ns)¹⁸ which are approximately the same as most ICFR irradiation conditions (see Table 2-1). There are also facilities which permit simultaneous introduction of helium gas via a second ion accelerator.¹⁸ Consequently, such an installation could be utilized to compare pulsed irradiation damage microstructures with microstructures developed during steady-state irradiations to the same total dose. With damage rates that are comparable to the midrange of those

anticipated in an ICFR and duplication of the expected pulse cycling, ion bombardment will be best used for low dose studies, since damage will accumulate at the same rate as that expected in-reactor. The testing could be accelerated by shortening the annealing time between pulses and increasing the pulse rate from 10 s^{-1} to, for example, 10^3 s^{-1} . Care must be taken, however, to demonstrate that this acceleration does not significantly change the characteristics of the point-defect behavior or the mechanisms governing the microstructural changes. The results from low-dose ion bombardment studies can be used to benchmark theoretical model development, and to verify whether bulk irradiation damage introduced by FFTF and/or WNR is representative of pulsed, high damage rate microstructures.

Bulk irradiation damage could be introduced by a stepwise process. The defect microstructure arising from pulsed fast neutrons could be nucleated by low-dose neutron irradiations in the WNR facility. Subsequently, the irradiation in a fission reactor such as the FFTF could be used to develop the microstructure and attain high doses. In addition, specimens may be removed from FFTF and re-irradiated in WNR to evaluate the iterative effects of pulsed irradiations upon various representative stages of developed damage microstructures. A qualitative sampling of radiation damage trends throughout a broad range of dose and defect structures could thus be obtained, and the validity of the defect structures produced by fission reactor and/or WNR irradiations could be evaluated by comparison with those obtained in ion experiments. In this manner we may be able to obtain an "effective" ICFR irradiation damage microstructure in specimens irradiated in FFTF and/or WNR. This step-wise irradiation would show the dominant effects of pulsed, high displacement rates upon the microstructure and provide specimens for mechanical properties tests. This method could be used to narrow the range of test materials and test conditions and indicate those experiments which should be performed under precise conditions for longer time periods.

Other possible radiation damage simulation facilities include high voltage electron microscopes (HVEM) and light ion accelerators. The HVEM can give a reasonably high displacement dose rate, $10^{-2} \text{ dpa-s}^{-1}$, but the Frenkel pair damage differs substantially from the cascade damage of 14 MeV neutrons. Another method which may be useful is bombardment with light ions, but dose rates are substantially lower ($\sim 10^{-3} \text{ dpa-s}^{-1}$ maximum) than desired unless the

ion energy is extremely high (like the 800 MeV protons at the WNR); however, the radiation damage recoil spectrum differs from that for 14 MeV neutrons. Overall these alternative methods do not appear as attractive as the simulations previously discussed.

The alternative to the use of currently available irradiation simulation techniques is to prepare a dedicated materials test facility. This may prove to be a necessary step if pulsed irradiation produces unpredictable and dramatic effects on the microstructure of test specimens. A feasible approach to this procedure could be the adaptation of one of the current ICFR developmental facilities (LASL laser project, LLL laser facility, or Sandia electron beam project).

The use of a dedicated test facility is, of course, contingent upon improvement of the pellet gain. Further, the minimum repetition rate of 10 s^{-1} must also be attained if prototypic ICFR conditions are to be realized; this requires both high reliability of the driver system and advances in pellet injection techniques. A high plant factor will be particularly important for a dedicated facility, since its primary function would be to produce bulk damage specimens for mechanical properties tests. Ideally this requires the attainment of fluences on the order of the anticipated lifetime fluence. The minimum goal fluence should at least be that attained in a year's operation (see Table 1-1 of the Introduction). Herein lie the major drawbacks of relying solely on the concept of a dedicated test facility. It is imperative that materials testing commence by 1982 in order to provide the requisite information for the EPR design engineers. Important decisions directly concerned with materials must be made in 1986 for ETF and in 1997 for EPR. Because of the proximity of these decision dates, the most reasonable approach would be to begin materials testing with currently available accelerators, pulsed neutron sources, and fission reactors. During the course of these simulation experiments, the importance of pulsed irradiation effects should be evaluated as soon as possible, and the feasibility of the dedicated test facility idea should be re-evaluated periodically. If quantitative testing becomes important due to unpredictably large pulsed irradiation effects upon the microstructure, the dedicated test facility must be built. However, until this need is demonstrated, the pressing time schedule requires that a materials testing program proceed with currently available irradiation testing facilities.

2.3.2 Experimental Program

In accordance with the approach suggested in the Introduction, the proposed program for studying microstructural changes in the ICFR environment emphasizes the ferritic and austenitic stainless steels for near-term experimentation. Scoping studies that examine the effect of pulsed irradiation on microstructural components (e.g. precipitates, voids, dislocations) and compare microstructures obtained under pulsed and steady-state conditions are scheduled early, in order to provide information for the decision on a dedicated materials test facility. Low-dose experiments using ion accelerators are suggested for initial efforts, because of their immediate availability. Studies of damage rate and pulse duration effects on microstructures may be most important because of the potential for accelerated testing either by decreasing annealing times or increasing the damage rate. The utility of data on pulsed effects obtained in the magnetic fusion energy (MFE) program can also be evaluated with the information gained.

The study of synergistic effects between displacement damage and helium, hydrogen and transmutant effects, and the preparation of mechanical properties test specimens by WNR and/or FFTF irradiations are scheduled later in the program, while the MFE program is completing studies in these areas¹⁰ so that full benefit may be derived from their effort.

Large scale testing of advanced alloys should be deferred until the EPR is available as the major test facility.

The suggested experimental program must be supported by a theoretical modeling effort. Much of this is of a fundamental level appropriate for Division of Basic Energy Sciences (BES) support, and is relevant to understanding the effects of both magnetic and inertial fusion irradiation environments. For example, both experimental and theoretical evidence shows that helium can strongly affect microstructural evolution; a knowledge of its properties and behavior in candidate materials is therefore essential. Specific items of information that are important include its production rate and diffusion properties; its interaction with other microstructural components such as dislocations, grain boundaries, cavities, dispersed phases, and specific to ICFR concerns, the dense cascade structure present at high damage rates; and its redistribution under cyclic temperature and stress conditions. Hydrogen production, diffusion, and distribution may be important along with its chemical

behavior. Data on high-neutron energy cross sections should be obtained. Production rates and migration characteristics of solid transmutants, their precipitation or redistribution behavior under cyclic conditions, and their influence on other elements of the microstructure are needed. Vacancy and interstitial production rates and production efficiencies as a function of damage rate (equivalently cascade density), migration under pulsed conditions, interactions with helium, hydrogen, and impurities are some of the point defect properties that need to be measured. Characterizing the primary damage state and dense cascade structure and development at high damage rates will be important.

Once the basic processes and parameters are understood, they can be incorporated into existing models for void and loop nucleation and growth, solute segregation, helium bubble formation and growth, precipitate coarsening, etc. These models are limited in scope, but can be used to estimate magnitudes of changes in individual components of the microstructure. Surface energies, defect and solute migration energies, defect binding energies, elastic energies associated with extended structures such as precipitates, coarsening and phase transformation kinetics, and thermodynamic properties of the candidate alloy systems are included in these physical models as parameters. In some cases the development of a microstructural feature depends strongly on a parameter: for example, void nucleation rate on surface energy or solute segregation on interstitial solute binding energy; it will then be important to have accurate values. The validity of currently available models under ICFR conditions will have to be determined; this is an area of concern because the neutron pulse times are short compared with the time scales of many physical processes. Unless it becomes clear that basic assumptions of the theories may be invalid under ICFR conditions, the major development of physical models can be left to BES and MFE programs. In order to study interactions among different microstructural elements, many investigators are developing comprehensive models that include several phenomena. An example is the inclusion of mobile helium in void nucleation and swelling theories, another is the simultaneous treatment of interstitial loop and void nucleation and growth. To obtain maximum benefit from the analytical work, experimental measurements of microstructural changes in well-characterized environments that are in some controlled way representative of the ICFR environment need to be compared with theoretical predictions. The microstructural data required include dislocation densities and structures, precipitate size distributions,

interstitial loop densities and size distributions, gas distributions, bubble and void sizes and number densities.

If the development of microstructural elements under controlled conditions is understood, comprehensive modeling can be used to assess interactions between the different elements and to extend the available data base to ICFR conditions. With a correlation between microstructure and mechanical properties, an assessment of first-wall response can be made.

Tables 2-2, 2-3, and 2-4 contain schedules and manpower requirements for investigations aimed at determining the magnitude of pulsed neutron effects on microstructure, evaluating the interaction of helium and displacement damage in a pulsed irradiation environment, and studying the relationship of microstructure and mechanical properties. The focus of the last effort is on mechanisms -- testing, for example, theoretical and model predictions of inreactor creep rates or confirming the predicted dependence of mechanical properties on various elements of microstructure. Programs for accumulating design data are discussed in the section on fatigue and mechanical properties.

Table 2-2.

Material: Ferritic and Austenitic Stainless Steels

Objective: To determine the magnitude of pulsed neutron effects on microstructure and compare microstructures obtained under steady-state and pulsed ion bombardment conditions.

Scope: The effects of damage rate, dose, and pulse duration on the development of key microstructural components will be investigated.

	Man Years											
	FY 1980-1995	1980		1987	ETF	1995	1997	EPR	2005	2005	Demo	2015
<hr/>												
A. Specimen selection	40		◄—►									
B. Test matrix development			◄—►									
(a) damage rate effects												
(b) dose effects												
(c) pulse duration effects												
C. Irradiations				◄—	○	●—	—	—	—	—	—	►
D. TEM examinations				◄—	○	●—	—	—	—	—	—	►
E. Analysis and modeling				◄—	○	●—	—	—	—	—	—	►
○	Decision date on Materials Test Facility											
●	Anticipated MTF operation, if MTF warranted.											

Table 2-3.

Material: Ferritic and Austenitic Stainless Steels

Objective: To evaluate synergistic effects of helium and displacement damage under ICFR pulsed conditions using dual-beam ion bombardment.

Scope: The effects of simultaneously produced helium key microstructural components will be investigated.

Table 2-4.

Material: Ferritic and Austenitic Stainless Steels

Objective: To obtain mechanical properties data from specimens irradiated in WNR and/or FFTF.

Scope: Bulk damage specimens will be prepared by irradiations in WNR and FFTF with validity of microstructures evaluated using ion bombardment data; mechanical properties data will be obtained.

	Man Years		ETF	EPR	Demo
	FY 1980 - 1995	1980	1987 1995	1997 2005	2005 2015
A. Irradiation technique development	40	◄►			
B. Validation of microstructures TEM examination		◄►			
C. Irradiations - sample preparation			◄►		
D. Post-irradiation testing			◄►		
E. Microstructure and mechanical properties correlation			◄►		

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3. FATIGUE AND MECHANICAL PROPERTIES

3.1 Scope

Fatigue is likely to be the predominant failure mode in all the proposed inertial confinement fusion (ICF) systems because of the pulsed nature of operation. The present program plan is intended to meet the design need based on fatigue and other mechanical criteria. It is recognized that the degradation of allowable material fatigue properties in an ICF environment occurs by corrosion, erosion and mass transfer, irradiation damage, helium and hydrogen effects. In particular, emphasis is made to insure that the design and operation requirements of EPR are adequately fulfilled.

At present, the proposed designs are still in the conceptual stage. The details of the component configuration and design are not yet established. However, as mentioned earlier, the choice of the structural materials for near-term ICF systems can be confined to the ferritic and austenitic stainless steels. This allows the definition and formulation of a laboratory test program with test conditions based on the information available from the various ongoing design efforts. In order to confirm the adequacy of the mechanical design, the present program includes component testing, which is expected to be accomplished by means of simulation experiments, and also experiments to be performed in ETF. It should be emphasized that the proposed program is not restrictive but is flexible enough to allow timely incorporation of new design considerations and experimental results into the test matrix.

Tensile, load relaxation, creep, creep-to-rupture and fracture toughness data on the structural materials are also required for the first wall structural design. For many of the ferritic steels and austenitic stainless steels of interest, these basic mechanical properties are available from LMFBR and MFR programs.^{1,2} Additional work is probably not needed except in the area of fracture toughness. Ferritic steels are known to suffer from temper embrittlement due to thermal aging, and also from the embrittling effects of irradiation, and helium, hydrogen and lithium exposure. The extent of embrittlement resulting from these causes in an ICFR can be different from that in an MFR and should be properly evaluated.

3.2 Problem Definition and Analysis

In this section, the mechanical, thermal, chemical and irradiation environments of the ICF systems to which the structural materials are exposed will be examined to form a basis for defining the design data requirements.

3.2.1 Mechanical and Thermal Environments

Recently, several investigators³⁻⁷ have addressed the sources of loadings for some ICF design concepts and have made approximate estimates concerning the magnitude of the stresses. Their results will be reviewed as a part of the background.

Fuel pellet microexplosion and blanket thermal expansion generate various mechanical loads on the first wall structure. In addition, spatial temperature gradients inside the first wall structure generate thermal loads. As mentioned earlier, the proposed program assumes that the first wall will be protected by various schemes, e.g., wetted wall, lithium fall or gas protected first wall. The sources of load will consequently be different depending on the first wall protection scheme. In the following sections, the sources and approximate magnitudes of the mechanical and thermal loads will be considered separately.

3.2.1.1 Mechanical Loads

For the wetted wall type of design, Bohachevsky and co-workers^{3,4} have estimated the magnitude of various mechanical loads. The conversion of debris kinetic energy into pressure on the inner first wall surface is a source of mechanical load for background gas densities below 10^{13} - 10^{14} atoms/cm³. A typical value for this type of load is a fraction of an atmosphere. When the ambient density in the cavity exceeds 10^{13} - 10^{14} atoms/cm³, the pellet explosion will generate a blast wave. For a total fuel pellet energy yield of 500 MJ with 20% of the total yield retained in the debris, the blast wave will produce a pressure of approximately 20 atm on the wall assuming a cavity radius of 5m. A most significant source of load is that produced by ablation-evaporation recoil. The ablation and evaporation of lithium will occur

following the absorption of X-ray pulse and plasma debris. The latter effect will be less important because the pulse duration for the debris is much longer than that for the X-rays. For a total pellet yield of 500 MJ with 5% of the total yield in the form of 1 μ second pulse of X-rays, the estimated pressure from ablation is approximately 60 atm at a 5m radius for a lithium-wetted wall. For the wetted-wall design, the thermal expansion of blanket material (lithium) will also exert a pressure on the channel walls following the deposition of the neutron-energy in the blanket. It is reasonable to assume that some type of channel structure will be required for lithium flow as a part of the first wall. The propagation and reflection of pressure waves will also produce loads on the channel walls. Bohachevsky and co-workers have estimated the lithium pressure produced by thermal expansion for the case where lithium is contained in two concentric spherical shells. For the inner shell, the calculated pressure is roughly 10 atm and is compressive.

With the estimated mechanical loads for the appropriate first-wall design, the static stress in the first wall can be calculated. For example, for a thin spherical shell, the membrane stress in the shell is $RP/2S$ where S is the thickness, P is the internal pressure, and R is the cavity radius. Based on the values of load given previously, the membrane stress in the thin spherical shell will in general be below the yield stress. Bohachevsky and co-workers^{3,4} and other investigators⁵ have pointed out that since the pressure pulse duration is extremely short (10^{-4} - 10^{-5} s) the stress in the first wall should be determined from an analysis of dynamic response. These investigators have performed calculations for a thin spherical shell subjected to an internal pressure pulse. Their results suggest that the amplitude of the stress wave is a fraction of the static values.

For the first wall protected by a lithium fall, the mechanical loads discussed earlier will not be applicable. Glenn and Young⁵ have estimated the pressure exerted on the first wall produced by the disassembly of an annular lithium fall. The pressure pulse consists of a strong pulse of approximately 150 MPa of a few μ s duration which is followed by a weak pulse with a duration of approximately 100 μ s. For a cylindrical reactor of a radius of 2-6m, the

maximum hoop stress (as calculated)⁵ is ~60 MPa which occurs for approximately 800 μ s after the initial impact. The stress in the first wall protected by a lithium foil will therefore also be below the yield stress of the wall material.

3.2.1.2 Thermal Loads

Static thermal loading will be produced by the steady state temperature gradient in the first wall structure. At a mean temperature of about 500°C for the first wall, it is anticipated that the resulting thermal stress will relax as a function of time. The magnitude of the thermal stress and its rate of relaxation have been examined.⁸ It can be assumed that with proper design, static thermal loading will not produce thermal stresses exceeding the yield stress.

Thermal stress pulses can be produced by the rapid application of thermal loads. When the energy of pellet explosion is deposited near a solid surface at a rate exceeding the rate at which heat can be conducted away into the interior, one would expect the generation of thermal stress waves. The magnitudes of this type of stress wave have been calculated.^{3,4,6} For the interest of the proposed program, the presence of the lithium layer at the wetted wall will protect the solid surface so that the thermal stress waves may not be an important consideration.

For an experimental reactor, the frequency of start-up and shut-down will be higher than that anticipated in a commercial plant. During the start-up and shut-down, transient thermal stresses can be significantly larger than the stresses discussed earlier. If the structural material is embrittled by irradiation, helium, hydrogen or temper embrittlement, one can expect structural failure produced by a brittle fracture mode. In the proposed program, the measurement of fracture toughness will be included. It is important that the fracture toughness of the first wall materials be properly evaluated as a function of the anticipated service history so that a useful basis for defining the mechanical design limits can be developed.

It is important to point out that the above considerations are valid for a simple solid wall and do not include complexities that arise from the presence of brazes, welds, joints and composite structures present in real

components. Complexities may also arise from the need for using optical ports and other problems associated with driver systems. Consideration should also be given to the transient thermal stresses associated with start-up/shut-down cycles, which are expected to be more frequent in EPR than in demonstration plants. Therefore, in the definition of data requirements, an allowance is made to insure that adequate information will be obtained so that it is applicable to a range of anticipated situations.

3.2.2 Chemical and Irradiation Environments

The composition and microstructure of the first wall structural materials are expected to be influenced by the chemical and irradiation environments present in the ICF systems, which will lead to significant degradation of the mechanical properties. The embrittlement of structural materials by neutron irradiation has been investigated in detail. However, the fatigue behavior as influenced by neutron irradiation as well as corrosion effects associated with lithium exposure are not well established. The effects of lithium exposure on structural materials may involve several mechanisms. It can simply be a surface effect which leads to corrosion fatigue or it can alter the bulk composition and microstructure which may result in the deterioration of fatigue resistance. In the proposed program, the above effects will be investigated in a systematic manner. The characterization of the appropriate chemical and neutron irradiation environments have been described elsewhere in this report.

3.3 Data Requirements

3.3.1 Fatigue

Two types of fatigue design data are required in the Materials Development Plan. The first type is obtained from laboratory experiments conducted on test specimens with simple geometries. The second type of data is generated by using actual or simulated components to characterize synergistic effects of various environments as well as to test the validity of design practice. The following section describes the details of the data requirements.

3.3.1.1 Laboratory Test Data

Since the first wall structure of an ICFR will be subjected to cyclic loading and will experience cycles $>10^8$ during its life-time, there exists a potential for damage resulting from high cycle fatigue. Therefore, experiments and analyses are required to demonstrate that the structural materials will be strong with respect to fatigue loadings in the appropriate environment and that fatigue crack propagation will not lead to premature failure of the first wall.

Laboratory test data will be needed for a wide range of structural materials that will be characterized by fatigue behavior, fatigue limits, crack initiation and crack propagation behavior as influenced by surface damage and environments encountered in different ICF conceptual designs. These studies should be undertaken to include the effects of several variables (e.g., temperature, cyclic frequency, neutron irradiation (both steady and pulsed) and surrounding environment (lithium) and helium and hydrogen effects). In addition, the effects of a number of metallurgical variables (e.g., heat-to-heat variations, thermal aging, grain size, etc.) on fatigue life and crack initiation and crack propagation behavior should be determined.

In the program prior to the operation of EPR, the materials of interest are ferritic steels and austenitic stainless steels. For a SOLASE-type concept, some fatigue tests will be performed on graphite. In the post-EPR program, it is anticipated that a wide range of materials currently being considered in the alloy development program will be incorporated.

The cyclic properties of the first wall materials in the laboratory can be presented in the following form:

- (i) $S - N_f$ curves (S = stress, N_f = cycles to failure)
- (ii) $S - N_o/N_f$ curves (N_o/N_f = fraction of life spent in crack initiation)
- (iii) $da/dN - \Delta K$ curves (da/dN = crack growth rate and K = stress intensity).

The thermal fatigue properties may be represented in the form of $\Delta T - N_o$ curves for the various geometries and materials where ΔT is the temperature range of cycling and N_o is the crack initiation life defined in a specific manner.

Fatigue life can be established under a variety of loadings (in a load- or strain-control mode) that are of interest to ICF reactors, and linear-elastic fracture mechanics techniques can be used to study crack propagation behavior. Investigations concerned with fatigue limits of laboratory specimens should address fatigue behavior based on a distinction between crack initiation and crack propagation, especially if the fatigue life is dictated by crack-initiation considerations as in the case of high-cycle fatigue. Even under low-cycle fatigue conditions, the per cent of life spent in initiating a crack length of about 0.1 mm in smooth laboratory specimens can be as large as 50-90% depending upon the strain range of cycling⁹ but this can decrease significantly if surface roughness is introduced.¹⁰ It should be noted that the real components of the ICF systems may not be smooth and in addition the first wall materials might experience erosion and pitting by contact with liquid lithium or lithium impact. Inasmuch as high-cycle fatigue behavior is extremely sensitive to surface roughness¹¹ and surface-integrity, the effects of surface roughness and other forms of surface damage (introduced either during machining or under in-service conditions) would be an important consideration in the design of the first wall, if it is to last as long as 30 years. It is also likely that both the chemical (lithium) and neutron irradiation environments affect the high-cycle fatigue life through the effects on crack initiation behavior. For example, neutron irradiation may impart resistance to high-cycle fatigue failure predominantly by delaying the initiation of cracks even though the hardening and embrittlement associated with neutron irradiation are expected to increase the crack-propagation rate. It is also important to point out that the fatigue properties derived from isothermal loading may not be directly applicable to thermal fatigue. One of the possible ways of investigating the thermal fatigue resistance of materials is by the "Glenny disk" method.¹² The ranges of the important test variables are:

Temperature: In general, fatigue life decreases and fatigue crack growth rates increase with increasing temperature. The magnitudes of these temperature effects should be quantified in the temperature range from 550-900° K.

Cyclic Frequency: In the several ICF conceptual designs, under normal operating conditions, it is expected that the frequency of cycling will be

1-20 cycles/s. Superimposed upon the main cycle of operation, there could occur a number of secondary waves which may lead to additional detrimental effects. For a commercial fusion reactor, the cyclic frequency can vary from the above high frequency range (normal operating conditions) to low frequencies (those associated with the reactor start-up/shut-down cycles). Some of the transient thermal stresses could lead to brittle fracture. In a lithium environment which is corrosive, cyclic frequency can have a significant effect upon fatigue life and crack initiation and crack propagation behavior. It is, therefore, important to characterize the effect of frequency of loadings in the appropriate environment applicable to various ICF reactor concepts. In addition, the loading waveform is a mixed spectrum of several frequencies and therefore should be an important consideration.

Stress Ratio (Mean-Stress Effects): The first-wall structural materials in the several ICF design concepts will experience loadings having a non-zero stress ratio, which can have a significant effect upon fatigue life and crack propagation rates especially if the stress ratio is greater than -1 (tensile mean stress). During its life-time, the first wall will experience loadings from different events representing many stress ratios and it is important that the effect of mean stress be known and a method be available for including the effects of mean stress for use in structural analysis.

Surface Damage: As discussed previously, surface damage is an important consideration in high-cycle fatigue and therefore experiments on specimens with controlled surface damage are needed. Surface damage can be introduced in a controlled manner by low-energy ion sputtering, lithium contact, or impact machining and grinding. Composition and microstructural changes that may occur in the surface region associated with the introduction of surface damage should be investigated.

Multiaxiality: The actual vessels are either spherical or cylindrical, and the state of stress is multiaxial. It is recommended that some biaxial fatigue tests be performed.

Helium Effects: The effects of different helium contents and its distribution, (produced by either tritium decay or thermal neutron irradiation), on high cycle fatigue life and fatigue crack propagation will be investigated.

Hydrogen Effects: The effects of hydrogen on high-cycle fatigue life and crack propagation will be investigated by performing the appropriate tests at various hydrogen pressures.

Lithium Effects: Two types of experiments will be performed to evaluate the effects of lithium on cyclic properties: post-exposure and in-situ experiments. In both types of experiments, the impurity content in lithium will be controlled to simulate the ICF conditions.

Irradiation Effects: The irradiation effects will be investigated mainly through the postirradiation tests conducted on specimens with microstructures (typical of those expected in ICF environments) produced by different techniques outlined elsewhere in the report. With the availability of ETF and EPR facilities, in-situ experiments will be performed as a part of the proposed Materials Development Plan.

3.3.1.2 Component Testing

It is well known that the dominant mode of failure in a modern aircraft is high-cycle fatigue and that the success of the large commercial airliner as an engineering system is heavily dependent on component testing as well as laboratory specimen testing. Therefore, component testing should be an important part of the Materials Development and Testing Program Plan. The component testing can be performed on several components at various levels ranging from small parts to full scale structures and can include synergistic effects of mechanical loads and materials variables such as weldments. This is a necessary proof test for the design practice to ensure high reliability of the system.

The first wall of an ICFR will be a composite structure including channels and supports required for lithium flow and for the various protection schemes. With the anticipated complexity, the first wall can be envisioned to be equivalent to the wing structure of an aircraft. Therefore, it will be difficult to develop and construct an EPR which is expected to have service life longer than 10 years without taking advantage of the engineering practice of the aircraft industry.

It is not straightforward to simulate the ICF environment and thermal loads in the laboratory. However, stress analysis and other relevant design calculations can be performed to estimate the magnitude and the frequency

of the various mechanical loads. Therefore, an important part of the proposed program plan will be the development of laboratory capability to simulate the required mechanical loads and to perform the appropriate tests on the various sections of the first wall in a systematic manner. The duration of some of these tests can be as long as several years. With the availability of an ETF, component testing can be conducted in a realistic environment. It is believed that a proper combination of laboratory specimen tests and component testing will ensure the successful design and operation of an EPR.

3.3.2 Other Mechanical Properties

The important sources of stress in the first wall structure, as discussed in the previous section, are static thermal loads produced by the steady-state temperature gradient and transient thermal loads produced by reactor start-up and shut-down and other temperature fluctuations. These thermal loads can be relaxed by inelastic deformation in the structural material. The rate of load relaxation, which in an ICFR will be influenced by irradiation-enhanced inelastic deformation (creep), is required in order to estimate the variation of stress level as a function of time. Tensile data are required so that the design stress can be specified below the yield stress. For thin section components under a stable load, the creep-to-rupture data are used to define design limits.

Tensile data are required for a temperature range of from room temperature to 873°K, a strain rate range of 10^{-3} to 10^{-6} sec $^{-1}$, and for both irradiated and unirradiated specimens. The required creep and load relaxation data will be for a temperature range 573 to 873°K, for a strain rate range of 10^{-3} to 10^{-9} sec $^{-1}$, and also for irradiated and unirradiated specimens. The creep-to-rupture data required will be for the same temperature range with a range of testing times up to 10^4 hours and for irradiated and unirradiated specimens.

The fracture toughness should be evaluated by charpy impact testing and by fracture mechanics methods. The temperature range of interest is from below room temperature to 873°K with specimens embrittled by thermal aging, irradiation, helium, hydrogen, and lithium exposure.

3.4 Strategy

From the viewpoint of reducing cost and time, it is necessary to take advantage of other available sources of test data, and laboratory and component testing capabilities. Test data will be generated by the LMFBR and magnetic fusion and energy sciences programs. These data may not be directly applicable to the ICF systems but they can be used to expand the data base to be generated in the proposed program so that improved capabilities for the correlation and extrapolation of the data can be developed. The laboratory testing capabilities developed in the LMFBR programs for fatigue testing in sodium and neutron irradiation environments as well as component-testing capabilities developed in the aircraft industry are a few examples of established technology which can be fully utilized in the present program.

The required design data on basic mechanical properties of ferritic steels and austenitic stainless steels are expected to be available from LMFBR and MFR programs. Some post-irradiation tests and in-reactor load relaxation and creep tests may be required if the irradiated microstructures produced in ICFRs are proven to be significantly different from those produced, for example, in MFRs. The heat treatment conditions for ferritic steels for ICFR application may not be the same as those for MFRs or LMFBRs and may lead to changes in the basic mechanical properties. Some mechanical testing on unirradiated materials will therefore be required. When evaluating these mechanical properties, advantage should be taken of recent developments in the characterization of inelastic deformation in order to save effort and to facilitate data correlation and extrapolation.¹³

The fracture toughness of a material is known to be sensitive to temperature and to the sectional thickness of the specimen. Presently available methods based on fracture mechanics considerations are not yet adequate for predicting, from laboratory data, the fracture toughness of a material the sectional thickness of which is different from that of the laboratory specimen. It is important in the laboratory work, therefore, to use specimens of thickness equivalent to that of the components of the first wall structure whenever possible. The dimensions of irradiated specimens and lithium exposed specimens may be limited by experimental space considerations, and the development of correlations for fracture toughness evaluation as a function of sectional thickness will be important.

Basic mechanical properties for graphite are also required. It is expected that much of the design data will be available from MFR programs.

The development of the Materials Program Plan will provide timely input into DOE supported design efforts. As the design effort progresses, various alternate design concepts will evolve and these changes could have a significant impact on the test conditions and the requirements of the proposed plan. One of the important considerations in the present plan is to keep close contact with the design efforts so that the relevant information concerning the design changes can be incorporated in the test program. In addition, important findings originating from the other efforts of the present plan, such as compatibility in liquid lithium and neutron irradiation studies, will be appropriately incorporated in the proposed program and used to adjust priorities. For a given material, an extensive data base will be generated in air and inert gas environments. From this data base, selected tests will be conducted in different environments to establish the effect of that environment on fatigue limits and fatigue crack initiation and propagation behavior. This approach will result in substantial savings of time and cost.

Table 3-1.

Material: Ferritic Steel and Austenitic Stainless Steel

Environment: Air and Inert Gas

Objective: To evaluate the fatigue and mechanical design data requirements for ETF and EPR.

Scope: The effects of a large number of important variables on fatigue limit, fatigue crack initiation and crack propagation in air and inert gas environments will be examined.

	Man Years		ETF	EPR	Demo
	FY 1980-1995	1980	1987 1995	1997 2005	2005 2015
A. Fatigue Limits	9				
(a) Specimen testing and development			←→		
(b) Temperature					
ETF temperature			←→		
Other temperatures				←→	
(c) Frequency and waveform					
ETF frequency and waveform			←→		
Other frequencies and waveforms				←→	

Table 3-1. (Continued)

		Man Years			ETF		EPR		Demo
		FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
(d) Mean stress effects			◀	=====	▶				
(e) Surface effects				◀	=====	▶			
(f) Multiaxial effects				◀	=====	▶			
(g) Metallurgical effects									
Heat-to-heat				◀	=====	▶			
Heat treatment		◀	=====	▶					
Weldments		◀	=====	▶					
B. Fatigue Crack Initiation		7							
(a) Techniques and development		◀	=====	▶					
(b) Temperature		◀	=====	▶					
(c) Frequency and waveform		◀	=====	▶					
(d) Surface effects		◀	=====	▶					
(e) Metallurgical variables		◀	=====	▶					

Table 3-1. (Continued)

		Man Years		ETF	EPR	Demo			
		FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
(f) Thermal fatigue
C. Crack Propagation		7							
(a) Development and techniques
(b) Temperature									
ETF temperature
Other temperatures
(c) Mean stress
(d) Metallurgical effects
D. Component		12							
(a) Development and techniques
(b) Testing

Table 3-2.

Material: Ferritic Steel and Austenitic Stainless Steel

Environment: Lithium

Objective: To evaluate the fatigue and mechanical design data requirements for ETF and EPR.

Scope: The effects of a large number of important variables on fatigue limit, fatigue crack initiation and crack propagation in lithium will be investigated.

	Man Years FY 1980-1995	1980	1987	ETF 1995	1997	EPR 2005	2005	Demo 2015
A. Post-exposure	6							
(a) Development and techniques			◀	▶				
(b) Fatigue limits			◀			▶		
(c) Crack initiation			◀			▶		
(d) Crack propagation			◀			▶		
(e) Composition and microstructure			◀			▶		
B. In Situ	6							
(a) Development and techniques			◀	▶				

Table 3-2. (Continued)

	Man Years							
	FY 1980-1995	1980	1987	ETF	1995	1997	EPR	2005
								Demo
								2015
(b) Fatigue limits	◄							►
(c) Fatigue crack initiation	◄							►
(d) Fatigue crack propagation	◄							►
(e) Composition and microstructure examination	◄							►

Table 3-3.

Material: Ferritic Steel and Austenitic Stainless Steel Containing Helium

Environment: Inert Gas

Objective: To evaluate the fatigue and mechanical design data requirements for ETF and EPR.

Scope: The effects of helium content and its distribution on fatigue limit, fatigue crack initiation and crack propagation in an inert gas environment will be examined.

		Man Years		ETF		EPR		Demo	
		FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
A.	Post-irradiation Tests	7							
	(a) Specimen preparation and techniques								
	(b) Microstructure characterization and helium distribution								
	(c) Fatigue limits								
	Development and techniques								
	Tests								
	(d) Fatigue crack initiation								

Table 3-3. (Continued)

	Man Years		ETF		EPR		Demo	
	FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
(e) Crack propagation								
Development and techniques
Tests

Table 3-4.

Material: Ferritic Steels and Austenitic Stainless Steel

Environment: Hydrogen

Objective: To evaluate the fatigue and mechanical design data requirements for ETF and EPR.

Scope: The effects of hydrogen on fatigue limit, fatigue crack initiation and crack propagation in hydrogen will be investigated.

		Man Years		ETF		EPR		Demo	
		FY 1980 - 1995	1980	1987	1995	1997	2005	2005	2015
A.	In Situ	7							
	(a) Construction and development of test facilities
	(b) Fatigue limits
	(c) Fatigue crack initiation
	(d) Fatigue crack propagation

Table 3-5.

Material: Ferritic Steels and Austenitic Stainless Steel

Environment: Neutron Irradiation

Objective: To evaluate the fatigue and mechanical design data requirements for ETF and EPR.

Scope: The effects of neutron irradiation on fatigue limits and fatigue crack propagation will be investigated by both post-irradiation and in situ experiments.

		Man Years		ETF		EPR		Demo	
		FY 1980 - 1995	1980	1987	1995	1997	2005	2005	2015
A.	Post-irradiation	8							
	(a) Develop techniques and facilities								
	(b) Fatigue limits								
	(c) Fatigue crack propagation								
B.	In Situ								
	(a) Develop techniques and facilities								
	(b) Fatigue crack propagation								

Table 3-6.

Material: Graphite

Environment: Inert Gas

Objective: To evaluate the fatigue and mechanical design data requirements for ETF and EPR.

Scope: The effects of a large number of variables on fatigue limit in an inert gas environment will be investigated.

Man Years		ETF		EPR		Demo	
FY 1980 - 1995	1980	1987	1995	1997	2005	2005	2015
A. Fatigue Limits	3						
(a) Specimen testing and development		←	→	←	→	←	→
(b) Temperature		←	→	←	→	←	→
(c) Frequency and waveform		←	→	←	→	←	→
(d) Mean stress effects		←	→	←	→	←	→

Table 3-7.

Materials: Advanced Alloys

Environments: Air, Inert Gas, Lithium and Neutron Irradiation

Objective: To evaluate the fatigue and mechanical design data requirements after EPR.

Scope: The effects of a large number of variables including hydrogen and helium on fatigue limits, fatigue crack initiation and crack propagation will be investigated.

	Man Years		ETF		EPR		Demo	
	FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
A. Fatigue Limits	2						←	→
B. Fatigue Crack Initiation	2						←	→
C. Fatigue Crack Growth	2						←	→
D. Multiaxial	2						←	→
E. Thermal Fatigue	2						←	→

Table 3-8.

Material: Ferritic Steels and Austenitic Steels

Environment: Air or Inert Gas

Objective: To evaluate mechanical properties design data for ETF and EPR.

Scope: The tensile, load relaxation, creep, creep-to-rupture and fracture toughness properties will be examined.

		Man Years			ETF		EPR		Demo
		FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
A.	Tensile Properties	2
B.	Creep and Load Relaxation Properties	3
C.	Creep-to-Rupture Properties	3
D.	Fracture Toughness Properties	6

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4. LITHIUM COMPATABILITY AND OTHER ENVIRONMENTAL EFFECTS

4.1 Scope

The utilization of liquid lithium in several ICFR design concepts is proposed for first wall protection.¹ In these design concepts liquid lithium is also used for tritium breeding and as the primary coolant. The compatibility of liquid lithium with various first wall, blanket and optical systems materials is therefore an important consideration for ICFR development.

As mentioned earlier, the choice of the structural materials for near term ICFR systems can be confined to the ferritic steels and austenitic stainless steels. Of these two types of material, more information on lithium compatibility is available for austenitic stainless steels. These data suggest potential problems in this area. For long term applications, the operating temperature for an austenitic stainless steel-lithium system appears to be restricted to below 500°C. Properties of ferritic steels may not be significantly superior to those of austenitic stainless steels. Since the former steels are strong candidates for structural materials, it is essential that efforts are made to qualify these steels for lithium compatibility and to accumulate sufficient information to specify operating conditions of near term ICFRs.

An important finding of the reported work is that ferrous alloys will suffer intergranular corrosion and penetration and that the rate of attack depends on the stress level, the composition of the lithium, and the composition and structure of the material.^{2,3,4} These observations suggest that the fracture and fatigue properties in lithium can be significantly different from those obtained during post-exposure testing, and in-situ experiments are needed. In this section, therefore, facility requirements for mechanical testing in lithium will be considered as well as the development of lithium handling capabilities.

There are design concepts for which liquid lithium will not be used.¹ Other environmental effects will exist and will be considered in this section.

4.2 Background

It is generally recognized that lithium is more corrosive toward ferrous alloys than sodium,^{2,3,4} and experience and data gained in the liquid metal fast breeder reactor program will therefore be of limited applicability. The various modes of lithium attack that have been identified include:

- 1) Simple solution
- 2) Intergranular penetration
- 3) Impurity reactions
- 4) Temperature gradient mass transfer
- 5) Concentration gradient mass transfer or dissimilar metal mass transfer.

The following variables have been found to influence the rate of attack in ferrous alloys:

- 1) Temperature
- 2) Temperature gradient
- 3) Cyclic temperature fluctuation
- 4) Purity of liquid lithium
- 5) Flow velocity
- 6) Number of materials in contact with the same liquid lithium
- 7) Composition and structure of the material
- 8) Stress state.

The attack by intergranular penetration is a phenomenon which requires special attention. This type of attack does not occur in the sodium environment and hence is not a problem in sodium technology. One can easily visualize that the grain boundary process involved will enhance crack initiation and propagation under an applied load, and lead to severe degradation of the mechanical strength of the material. A thorough investigation of this phenomenon is required to obtain information to be used for materials selection and reactor design.

Based on reported work,^{2,3,4} intergranular attack is caused by the fact that the free energy of formation of lithium carbide and nitride is lower

than that of some of the chromium carbides and nitrides. The difference in the free energy of formation provides the driving force for lithium to diffuse into the steel and to reduce chromium carbides and nitrides. It is well known, especially in ferritic steels, that different carbides of varied stability exist depending on the prior heat treatment and thermal aging conditions.⁵ The rates of intergranular attack in weld metals and in the heat affected zone can therefore be expected to differ from that in the base metal. The rate of attack is also likely to change with time during service due to thermal aging. The carbon and nitrogen content of the steels and the composition of major components should have a strong effect on the resistance to intergranular attack. Recently, irradiation-induced segregation and precipitation have been shown to occur.⁶ These processes could potentially form carbides of varied stability and thus have an indirect effect on the rate of attack.

A significant finding of the past work on intergranular lithium attack is that the rate of attack increases with increasing carbon and nitrogen content in lithium. This effect may be caused by the carbon and nitrogen diffusing along grain boundaries and forming chromium carbides and nitrides which are subsequently reduced by the lithium. The observed dependence suggests the importance of controlling the impurity level in lithium.

Based on free energy considerations, group IV, V, and VI elements in the periodic table can cause chemical effects similar to those produced by carbon and nitrogen described above. Pellet debris will contain elements of these groups and will enter lithium as impurities. In addition to the possibility of enhancing intergranular attack, some of these elements (e.g., S, Sb, P, etc.) may enter grain boundaries and produce embrittlement effects similar to those resulting from temper embrittlement. In the proposed program, these possibilities will be investigated so that potential problem areas can be identified and near term remedies sought if required.

The effects of stress on intergranular attack have been reported by several investigators.^{2,3} Under loading conditions where intergranular crack initiation and propagation occur, it will be difficult to separate stress-assisted intergranular attack from environment-enhanced crack initiation and growth, and in-lithium mechanical testing becomes important.

A part of lithium technology needed for the materials program is the capability of impurity removal and composition control.^{7,8} The techniques for continuous monitoring of the impurity level in the lithium loop are also required, since the importance of performing materials testing in well-controlled and characterized liquid lithium is recognized. It is expected that the development of the experimental capabilities for the proposed program will also serve the purpose of establishing the required lithium technology in the areas mentioned above.

Hydrogen is known to cause embrittlement at lower temperatures, to form hydrides in some metals, e.g., refractory alloys. It is also known to react with carbon in steel to form methane bubbles at grain boundaries. The latter reaction may be enhanced by irradiation directly and indirectly by irradiation-induced precipitation and segregation, which may form carbides of lower stability. Since similar problems will also occur in magnetic fusion materials, the investigation of hydrogen effects other than its influence on fatigue properties will not be included in the proposed program.

Mirror materials will be in contact with liquid lithium and lithium vapor and also with pellet debris. In the SOLASE concept⁹ liquid lithium is not used, but chemical reactions between graphite and unburned fuel and pellet debris may occur. The energy sink material, used in reactor concepts where the first wall is protected magnetically, will interact with charged particles and pellet debris. These are potential problem areas where materials development is required and will be included in the proposed program.

4.3 Data Requirements

4.3.1 Lithium Compatibility

Engineering data are required on the rate of various modes of lithium attack and their dependence on the variables given at the beginning of the section on background. Mechanistic studies are required to provide a basis for data correlation and extrapolation and for materials selection. In the following, the different modes of lithium attack will be discussed separately.

4.3.1.1 Solution Attack

The rate of solution attack can be measured by weight loss and dimensional changes. The rate of solution of the major elements in ferrous alloys is known to vary from one element to the other.² The characterization of the depth and the composition and structure of the affected surface region in contact with liquid lithium is also a necessary part of data collection, since the preferential removal of nickel has been shown to produce a ferrite surface layer in austenitic stainless steels. The experiment should be performed as a function of temperature, temperature cycle, flow rate and controlled concentration of non-metallic elements and other impurities in liquid lithium. Some measurements should be performed with irradiated specimens whose surface condition has been controlled during irradiation.

The solubilities of the major elements in ferrous alloys are important fundamental parameters that define the driving force for solution attack. Literature values should be confirmed under controlled experimental conditions and further tests conducted to determine the effects of impurities which are expected in an ICFR environment.

4.3.1.2 Intergranular Penetration

The possible mechanisms involved in intergranular lithium attack have been discussed in the background section. This is an area where extensive effort is expected. Controlled experiments are required to determine the rate of attack as a function of temperature, temperature cycle, flow rate, concentration of non-metallic elements and other impurities in lithium, and the stress state. The variations in composition and microstructure in the steels resulting from fabrication history, heat treatment, thermal aging and irradiation as discussed previously are expected to have a strong effect on the rate of attack. A systematic investigation of the influence of these materials variables should be made.

The rate of attack can be measured from the depth of penetration. The characterization of the corrosion product in the grain boundary and the related compositional and microstructural changes in the grain matrix are of major importance. Various advanced diagnostic tools are available today and should be used in the proposed program.

The effect of stress has been shown to be an important consideration. Only the in-lithium mechanical testing under static loading is considered in this section, since the in-lithium mechanical testing under cyclic loading is included in the section of the proposed program dealing with fatigue. It is expected that these experiments will yield information on the effects of stress on the rate of intergranular attack as well as data on the effect of lithium environment on creep, stress to rupture, and fatigue properties of the proposed structural materials.

4.3.1.3 Impurity Reactions

It is well established that interstitial elements such as carbon and nitrogen can enter or leave a ferrous alloy depending on the chemical potential of the particular element in lithium and in the metal. The resulting compositional and microstructural changes in the material can strongly influence its mechanical properties; for example, severe decarburization can significantly reduce the creep strength of the material. The rate of the reaction can be determined from the depth of the affected zone and from the resulting compositional and microstructural changes. Since long term in-lithium mechanical testing can be costly, post-exposure mechanical testing will be utilized whenever possible in the proposed program. Post-exposure testing can be justified for cases where mechanical properties are governed by the bulk properties and where the anticipated compositional and structural changes can be produced by proper lithium exposure treatment.

An important consideration for the impurity reactions is to predict the rate of reaction at long times, so that one can accurately calculate, for example, the depth of the decarburized zone and estimate the remaining mechanical strength. For this purpose it is essential that the thermodynamics data that govern the driving force for impurity reactions be well established. Experimental data are also needed on the kinetics constants such as the diffusion coefficient or the reaction rate constants for the rate determining processes. If literature data are not available, efforts are expected to be made to determine the values of these basic parameters.

Carbon and nitrogen in lithium are known to enhance the rate of intergranular attack. Other impurities, as discussed in the section on background, such as Pb, Sb, etc. may cause similar effects. These impurities may

also enter the grain boundaries and cause embrittlement. In addition to determining the rate of reactions of the latter impurities, it is important that the relevant thermodynamic and kinetic data be established for data correlation and extrapolation.

4.3.1.4 Temperature Gradient and Concentration Gradient Mass Transfer

Temperature gradient mass transfer is known to occur in lithium loops.² Concentration gradient mass transfer can occur between dissimilar materials and between weld metal and base metal. It is possible that temperature gradients and concentration gradients will co-exist, depending on reactor design. The mass transfer rate should be measured systematically as a function of temperature, temperature gradient, composition gradient, temperature cycle, flow rate, and the purity of liquid lithium. The rate can be determined from weight and dimensional changes. It is important, as for other modes of attack, that the surface region of the material be well characterized in terms of the depth of the affected zone, composition, and microstructure.

Rate equations based on thermodynamics and kinetics as well as on fluid dynamics considerations should be established from experimental data. Since the temperature and concentration gradients in a reactor can vary over a large range, the rate equations obtained will be useful for reactor design.

4.3.1.5 Materials Variables and Experimental Conditions

As pointed out previously, the initial phase of the materials program will emphasize ferritic steels with austenitic stainless steels as the alternative structural material. Several ferritic steels¹⁰ of varied chromium content are strong candidates and will be included in the proposed program. It is recognized that the properties of ferritic stainless steels are sensitive to heat treatment, fabrication history, thermal aging, and minor alloy additions such as carbide stabilizers. The effects of these materials variables will be investigated systematically. Consideration of weldments and the heat-affected zone will be included with the materials variables.

The various experimental conditions of interest can be summarized as follows:

Temperature:	573-873°K
Lithium Velocity:	0-8 m/sec
Stress Levels:	0-300 MPa
Temperature Difference:	0-200°K
Lithium Purity:	Max. nitrogen 1000 ppm. Max carbon 200 ppm and other impurity concentrations to be dictated by reactor design.
Concentration Gradient:	The choice of materials of interest will depend on reactor design.

4.3.2 Other Environmental Effects

Exploratory work to investigate the effects of ICFR environment on mirror and energy sink materials should be initiated. If problem areas exist, they may be solved in part by materials selection rather than by materials development. The interaction between graphite and unburned fuel and pellet debris can be simulated by ion bombardment and other techniques. Some of the same interactions will also occur in magnetic fusion reactors and the proposed program will utilize results of materials research designed for magnetic fusion reactor development.

4.4 Strategy

The lithium loop used in the test programs should have controlled nitrogen, carbon, oxygen, and hydrogen levels as well as other impurity concentration levels. The capability for continuous monitoring of these compositions is required.

Lithium loop tests need to be conducted with progressive complexity of experimental conditions so that the effects of a particular variable can be isolated. For test programs prior to EPR, lithium loops of small scale will be used. After EPR, some of these small loops can be used for long time testing. It is planned that large scale lithium loops will be constructed and be available after EPR. The large scale loops could actually be a part of

EPR. These large loops can be used to test the synergistic effects of several critical variables, to confirm the validity of the various rate equations developed for reactor design, and to proof-test the various impurity trapping and removal capabilities as well as the equipment for monitoring the impurity level.

Mechanical testing in-lithium is emphasized in the proposed program because of the influence of stress on corrosion. Post-exposure testing will be utilized to determine the effects of impurity reactions such as decarburization on the mechanical properties. Compositional and microstructural changes produced by long term lithium exposure with and without radiation will also be investigated.

An important requirement of lithium research is developing rate equations for lithium attack that are capable of predicting the long-time behavior of the material. For this purpose, mechanistic knowledge of various modes of attack is essential. The various advanced diagnostic tools available today should be fully utilized to characterize the composition and structure of corrosion products and the related compositional and microstructural changes in the material. The results of detailed chemical and physical analyses should aid considerably in understanding the mechanisms involved in lithium attack.

Table 4-1.

Material: Ferritic Steels and Austenitic Stainless Steels

Environment: Flowing Lithium (Small scale loops)

Objective: To evaluate lithium corrosion and mass-transfer in lithium.

Scope: The rate of various modes of lithium attack will be measured as a function of impurity levels, environmental and materials parameters, and rate equations for design applications will be developed.

		Man Years			ETF		EPR		Demo	
		FY 1980-1995	1980	1987	1995	1997	2005	2005	2015	
A.	Corrosion and Mass Transfer	15								
(a)	Lithium loop construction and development			◄	►					
(b)	Temperature and temperature cycle			◄	►					
(c)	Temperature and concentration gradient			◄	►					
(d)	Flow rate			◄	►					
(e)	Purity of lithium			◄					►	
(f)	Materials variables, composition and structure			◄					►	

Table 4-2.

Material: Ferritic Steels and Austenitic Stainless Steels

Environment: Flowing Lithium (Small scale loops)

Objective: To evaluate effects of lithium on mechanical properties.

Scope: The effects of stress state on corrosion rates and the effects of a lithium environment on mechanical properties (fatigue, etc.) of steels will be investigated.

	Man Years		ETF		EPR		Demo	
	FY 1980-1995	1980	1987	1995	1997	2005	2005	2015
Effects on Lithium on Mechanical Properties	24							
(a) Lithium loop construction and development			◄	►				
(b) Temperature and temperature cycle			◄	►				
(c) Flow rate			◄	►				
(d) Purity of lithium			◄	►				
(e) Materials variables, composition and structure			◄	►				
(f) Stress state (Fatigue, Creep)			◄	►				
• In lithium mechanical testing								
• Post-exposure mechanical testing								

Table 4-3.

Material: Ferritic Steels and Austenitic Stainless Steels

Environment: Flowing Lithium

Objective: To evaluate lithium compatibility design data for demonstration plant and beyond.

Scope: The rate of various modes of lithium attack will be measured as a function of environmental and materials parameters and rate equations for design applications will be developed.

	Man Years /yr. FY 1980-1995	1980	1987	ETF	1995	1997	EPR	2005	2005	Demo	2015
A. Loop construction and development	3										
B. Lithium attack measurements	0										
(a) Solution attack											
(b) Intergranular attack											
(c) Impurity reactions											
(d) Temperature and concentration gradient mass transfer											

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5. EXECUTIVE SUMMARY

The requirements of the inertial confinement fusion program generate issues of two types when materials development is considered. Technical questions arise because of unfamiliar environmental conditions such as pulsed irradiation, cyclic mechanical and thermal loadings, and extensive use of liquid lithium. Other issues arise because of the time frame for engineering development. In this plan, we attempt to balance both technical and scheduling concerns.

The engineering development schedule, shown in Fig. 5-1, calls for design decisions on the Engineering Test Facility (ETF) in 1987. Because of the nearness of this date, it is unlikely that any alloy system not already well characterized will be used in the construction of the ETF. The Experimental Power Reactor (EPR) will be the first large volume, high plant factor facility capable of generating materials data required for engineering design. Thus, the schedule for development of inertial confinement fusion suggests an approach to materials development that divides the time frame into two periods.

During the first time period, which will extend from the present until the date of initial EPR operation or about the year 2000, the development effort should concentrate on the qualifications of the ferritic stainless steels as reference structural materials and the austenitic stainless steels as alternates, with only a low level of activity on other alloy systems. The second time period extends from the date of initial EPR operation through the construction of the first commercial machine. During this period, large scale testing of other systems along with those previously selected should occur using the EPR itself as a major test bed. The justification for this strategy is based, in part, on the promise and potential of the ferritic stainless steels for fusion application. This class of commercially available steels appears to be resistant to void swelling and helium embrittlement, while being compatible with liquid lithium. Key questions about its applicability do remain, however, and must be addressed at an early stage in the program. One of the areas of concern is the compatibility of pellet impurities with the ferritic and austenitic stainless steels.

The first decision point in the program occurs in 1985, when a decision as to whether or not the ferritic stainlessness are appropriate for the construction of the ETF and remain promising for EPR should be made (see Fig. 5-2).

Fusion Development

(Reference Budget Case)

Fig 5-1. Schedule for fusion reactor development [Ref. J. M. Deutch, Department of Energy Policy for Fusion Energy, DOE/ER-0018, Sept. 1978.]

If they are not appropriate, then the following period can be spent in an intensified effort to qualify an austenitic stainless steel for ETF application. Otherwise, the two years prior to the ETF decision date will be used to select a specific ferritic alloy from the range of possibilities. It should be noted that much experience can be gained if the same material can be used for both the ETF and the EPR. Thus, if it becomes clear that the ferritics cannot be used at the higher dpa levels required by EPR, they should probably not be used in ETF.

A second major question in the materials development program is whether or not the waterfall concept can be engineered for the driver system most appropriate for commercial development. The incentive for developing the waterfall system is based on its amelioration of radiation damage and pressure pulse effects with the consequent extension of the structure lifetime. If the concept is not workable, the severity of the first wall environment increases, and the focus of the materials program may change. Thus, a two-pronged effort that closely couples chamber design activities with the material development effort is also suggested. This effort will then support the decisions on vessel design that must be made for ETF in about 1990 and for the EPR in about 1999.

If it can be shown that pulse effects are not significant, and that available alloys such as the ferritic stainless steels are capable of surviving neutron wall loadings of 40 or more MW-yr/m², then the additional engineering complexities of the waterfall approach may not be justified. A decision on the feasibility of the concept has been scheduled in the material development plan for 1985. If the waterfall design is proven, then decisions on its use in ETF and EPR blankets are indicated for 1988 and 1998.

A third major decision for the program is whether or not a dedicated materials test facility is needed to allow the confident construction of an EPR. This decision depends strongly on the nature of pulsed neutron damage and its effects on the microstructure and mechanical properties of materials. It is a conclusion of this assessment that currently available or planned accelerators and fission reactors can be used to assess the importance of pulsed effects on microstructural development and to determine their relative magnitude. Clearly if the effects are large, there will be a strong incentive to develop and operate a separate materials test facility. If not, then

1. Are Ferritic Stainless Steels Applicable?

1

High dose effects
High He effects
DBTT shift - Tougheners
Weld performance
High cycle fatigue

2. Are there Certain Pellet Impurities that are Incompatible with Using a Ferritic or Austenitic Structure?

2

3. Can a Waterfall Concept be Used?

3

- (a) ETF Blanket design picked
- (b) EPR Blanket design picked

3a

3b

4. Is a Dedicated Facility Needed to Build EPR?

4

5. What are the Magnitude of Pulsed Effects on Microstructure Development?

5

6. What are the Costs of a Low Gain (~ 25) High Duty Factor Material Test Facility?

6

7. Define Severity of Corrosion-Fatigue Problem in Li.

7

Figure 5-2. Key questions and decision rates in the Material Development Program for an inertial confinement fusion reactor.

it is felt that the EPR itself can serve as the central high volume test facility to qualify materials for the demonstration plant and the first commercial machines. The large scale development of more advanced alloy systems, such as those based on vanadium, should be delayed until the applicability of the ferritic stainless steels has been determined, and the EPR is available for use as an irradiation test bed.

As input to the decision on a materials test facility, an estimate of the costs and schedule for building a high duty factor, low gain materials test facility must be available by the 1986 decision date.

All of the inertial confinement designs will be subject to high cycle fatigue loading. It is important that testing of this key property area be conducted on candidate materials for the construction of the EPR. It will be particularly important to determine the effect of surface damage processes on the high cycle fatigue life-time.

The discussion above is summarized in Fig. 5-2, which lists key questions that the materials development program must address and gives dates when information should be available so that timely design decisions can be made. Answers to questions 1, 2, 5, and 7 depend directly on information that will be obtained from the technical programs discussed in this report; major tasks for these programs have been listed in Tables 5-1 to 5-4 along with schedules for their completion.

Table 5-1.

Question: Are ferritic stainless steels applicable for inertial confinement fusion reactors?

● Interim assessment report.

1. Select candidate ferritic alloys.
2. Plan test matrix for determining effects of irradiation, Li soaking, thermal aging on toughness (including DBTT) on both base and welded samples of different initial heat treatments.
3. Define effects of ICF environment on toughness sufficiently to allow construction decision on ETF.
4. Define welding practice for candidate ferritic steels.
5. Develop plan for determining the relationship between mechanical properties and microstructure as a function of dose up to 400 dpa using input from study on pulse effects.
6. Conduct irradiation program using fission reactors and accelerators to define mechanical properties as a function of dose.
7. Develop plan for defining the role of helium in the ferritic stainless steels.

Table 5-1. (Continued)

8. Test for resistance to helium embrittlement using cyclotron injected and nickel-containing neutron irradiated alloys.
9. Conduct high cycle fatigue testing including corrosion fatigue effects.

Table 5-2.

Question: Are there certain pellet impurities that are incompatible with using a ferritic or austenitic structure?

1. Select materials for experiments.
2. Plan test matrix to study effects of pellet impurities on lithium compatibility.
3. Perform experiments in lithium on the rate of solution attack, intergranular attack, impurity reactions and temperature gradient mass transfer with controlled impurity levels.
4. Perform mechanical testing to evaluate post exposure fracture toughness and fatigue properties.
5. Develop rate equations for lithium attack under the influence of pellet impurities.
6. Analyze results and produce preliminary recommendations.

Table 5-3.

Question: What are the magnitudes of pulsed effects on microstructural development?

1. Select materials for experiments.
2. Plan test matrix for study of damage rate and pulse duration effects on microstructural components.
3. Perform theoretical calculations with available models to predict magnitude of effects and analyze experimental results.
4. Conduct irradiation experiments in accelerators and fission reactors.
5. Analyze results.

Table 5-4.

Question: Define severity of corrosion fatigue problem in lithium.

1. Select materials for experiments.
2. Plan test matrix to evaluate corrosion fatigue properties of ferritic steels and austenitic stainless steels in lithium.
3. Perform experiments in lithium on the rate of solution attack, intergranular penetration, impurity reactions and temperature gradient mass transfer under controlled conditions.
4. Perform fatigue testing in lithium.
5. Perform post-exposure fatigue testing.
6. Develop rate equations for lithium attack.
7. Analyze results and produce preliminary recommendations.