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ALLOY DEVELOPMENT FOR IRRADIATION PERFORMANCE: PROGRAM STRATEGY*

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BASIS FOR ALLOY DEVELOPMENT

In recent studies[1,2] significant progress has been made in developing concepts for commercial fusion power reactors that would be economically competitive. One of the most critical and generic problem areas is that of structural materials for use in regions of high neutron flux (i.e., the first wall and blanket region). Excepting problems associated with the interaction of the first wall and the plasma, materials performance is generally an economic rather than a feasibility question. The properties of the structural material must allow a conservative design which will achieve temperatures, power densities, and component reliability and lifetimes that permit economically competitive power production.

The fusion reactor neutron spectrum is considerably more "damaging" in terms of displacement damage and transmutation products produced per neutron than is for example the fast fission reactor neutron spectrum. Table 1 compares damage production rates in two alloys, stainless steel and vanadium, in the two different spectra. The fusion reactor spectrum produces much larger quantities of helium than the fast reactor spectrum. If comparisons are made on a dpa or fluence basis, the higher helium production rates in a fusion spectrum result in more irradiation induced swelling and larger reductions in ductility properties. Parkins[3] suggests that the deleterious effects of irradiation on materials may limit component lifetime to the extent that the associated increase in cost may prevent the practical application of fusion energy.

TABLE 1. Damage Production Rates in Fusion Reactor and Fast Reactor (EBR-II) Spectra.

	Fusion reactor at 1 MW/m^2		EBR-II, Row 2	
	dpa/S 10^{-7}	appm He/S 10^{-7}	dpa/S 10^{-7}	appm He/S 10^{-7}
316 stainless steel	3.6	47	10.0	6.1
Vanadium	3.6	15	1.2	0.4

Recent experimental work[4,5] in which fusion reactor irradiation effects are approximated by mixed spectrum fission reactor irradiations (discussed in Technical Approach Section) supports the general conclusions that even in the fusion reactor environment the properties of an alloy system can be improved by the metallurgical variables of composition and structure. For example, Fig. 1 shows one important property, irradiation induced swelling, as a function of irradiation temperature in austenitic stainless steels. The irradiations were conducted in a mixed spectrum reactor resulting in helium concentrations and dpa levels equivalent to about 25 and 5 $MWyr/m^2$, respectively. In annealed 316, swelling is probably unacceptably high (8–10%) at all temperatures and increases sharply above 600°C. Controlling the structure by cold working markedly

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reduces the swelling at temperatures below 600°C. At higher temperatures the cold work is unstable and recovery and recrystallization occurs with concommitant increase in swelling. Changing the composition by the addition of a small amount of titanium (0.2–0.3%) reduces the swelling of annealed material for temperatures up to about 650°C. Composition and structure also influence mechanical properties,[4,5] differences stemming from changes in the accommodation of transmutation produced helium in the alloys. Although these results are for the austenitic stainless steels one would also expect to find beneficial effects of composition and structure in other alloy systems. There is thus a firm basis for beginning an alloy development program. The goals, strategy, and experimental approach of this program are outlined in this paper.

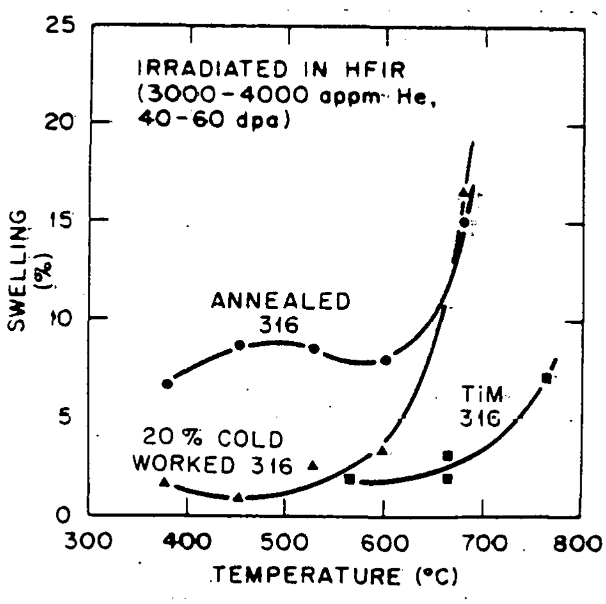

FIGURE 1. At dpa and Helium Levels Equal to About 5 and 25 MWyr/m^2, respectively, Fusion Reactor Exposure Control of Microstructure and Composition can be Utilized to Reduce Swelling.

GOALS AND ALLOY PERFORMANCE REQUIREMENTS

An alloy for use in fusion power reactors must have lifetime, power density and temperature capabilities which allow economically competitive generation of electricity. At this stage of fusion power reactor development the reactor designs are conceptual and evolutionary. It is not possible to be quantitative on lifetime and temperature requirements and thus one cannot place quantitative requirements on those mechanical and physical properties which impact lifetime and temperature capabilities. The required temperature capability depends on the heat conversion system being utilized and on details of the reactor design.

At minimum, any alloy should have temperature capability which allows reactors to be designed with a conventional liquid metal — H_2O steam generator similar to those used in breeder reactor systems. If a turbine-generator similar to those employed in light water reactors were used the fusion outlet coolant temperature (assuming a liquid metal coolant) would be about 475°C, metal temperatures in the first wall and blanket region would probably be in the range 325–525°C. However, by using available design options such as overcooling, temperatures in the region of high neutron flux could be reduced significantly.[6,7] The goal of higher operating temperatures to improve thermal efficiency and potentially reduce power costs is also reasonable. Alloys are included in the program which have temperature capability for use in a system having He-H_2O steam generators and direct helium turbines (metal temperature of about 730 and 900°C, respectively).

Economic modeling of fusion reactor systems shows that the required first wall lifetime is a function of plant thermal efficiency, wall loading, changeout time, additional costs due to materials requirements in other components, etc. The goal of the alloy development program is a material having a lifetime capability of 40 MWyr/m^2. It is realized that this lifetime may not be required for fusion power to be competitive and that the required lifetime will be different for each material because of temperature and wall loading capability, cost, etc.

Temperature and lifetime requirements, proposed breeding and heat transport fluids, fusion reactor power cycles, plant construction, and maintenance, combine to impose requirements on the alloy in several broad areas: availability, fabricability, corrosion and mass transfer, mechanical and physical properties, and degradation of these properties by neutron irradiation. Present industrial capacity would allow use of austenitic or ferritic steels, nickel base alloys, titanium alloys, or alunium alloys. For other alloy systems capacity would have to be developed or expanded. Although design specific, it is clear that fusion reactor first wall and blanket structures will require complex shapes and welded structures. Thus, the alloy must be readily fabricable and weldable. A number of reactor breeding and cooling fluids (lithium, molten salts, helium, water, and Li-Pb-compounds) have been proposed in various design studies. Any structural material must have acceptably low corrosion rates and be resistant to degradation of properties due to mass transfer with other materials in the reactor system through the cooling and breeding fluids. The presence of tritium in the system requires that the alloy be resistant to hydrogen embrittlement at anticipated operating and shutdown conditions. Mechanical and physical properties must be sufficient to allow a conservative design at the operating temperature and for the loading conditions required. Degradation of these properties below acceptable levels by neutron irradiation will define end of life. Nearly all common mechanical and physical properties directly affect design and thus alloy selection: thermal expansion, thermal conductivity, elastic modulus, tensile and creep strength and ductility, fatigue, fatigue-crack (flaw)-growth, fracture toughness. Additionally, the phenomena of irradiation induced swelling, phase instability, and creep which are functions of the irradiation environment are important. The relative importance of a given property is, however, design specific (e.g., the impact of fatigue and crack growth properties on alloy selection is strongly dependent on wall loading, burn time and metal temperature).

DEVELOPMENT SEQUENCE

Alloy development must proceed through sequential steps. First, one must determine the potential of a given alloy system for fusion reactor application. This can only be accomplished by examining alloy performance requirements and critical properties of the alloy system being evaluated. Scoping alloys are chosen to represent the different alloy types within a system (e.g., α, $\alpha + \beta$ and β titanium alloys from the titanium alloy system). The most important mechanical and physical properties as well as general effects of the fusion reactor environment (corrosion, mass transfer, irradiation effects) will be assessed using these scoping alloys. Information obtained on the scoping alloys will provide input to an analysis and evaluation effort having an objective of determining the potential of a given alloy system for fusion reactor applications. If a decision is made to proceed with an alloy system, a set of base research alloys representing those alloy types judged to have the greatest potential will be chosen. The main thrust of the effort on base research alloys will be to identify the compositional and microstructural variables which control critical properties and the response the alloy to the fusion environment. One or more prime candidate alloys are then selected which represent the alloy type within an alloy system having the most potential for meeting the fusion reactor needs. Final optimization is accomplished by control of microstructure and minor alloying element chemistry. The last step in the development sequence is determination of engineering properties. Extensive data will be required to develop constitutive equations which describe material behavior as a function of temperature, flux, stress, stress state, etc., for reactor design purposes.

PARALLEL PATH APPROACH

The diversity in requirements for a fusion reactor first wall and blanket structural material coupled with a limited amount of relevant data on which to base judgements forces the alloy development program to be broad in the range of materials considered. Premature selection or rejection of an alloy could severely limit design options at a later time. The development strategy thus consists of three paths (A. austenitic alloys, B. high strength Fe-Ni-Cr alloys, and C. reactive and refractory alloys) which represent those alloy systems judged to have the most inherent potential for fusion reactor applications and a fourth path (D. innovative material concepts) which will provide for development of new materials and concepts. Table 2 presents an evaluation of the critical properties of Path A, B, and C alloys for their use in fusion power reactors.

Path A — Austenitic Alloys

These alloys represent our most established technology. Production, fabrication, and welding are state-of-the-art. There is extensive information on phase stability, effects of composition on phase stability, and a more advanced understanding of irradiation effects than for any other system. The estimated upper temperature limit of 550°C is based on irradiation induced swelling and embrittlement observed in mixed spectrum reactor irradiations to helium contents equivalent to about 25 $MWyr/m^2$(4). The fatigue and crack growth properties, coupled with thermal expansion, conductivity, and modulus may limit first wall loadings to 2–3 MW/m^2(8) particularly for cyclic reactors such as Tokamaks. For the temperatures considered, these alloys have sufficiently low corrosion rates in lithium, could be used with a helium coolant, and are compatible with other alloys which would probably be used in the liquid metal – H_2O heat transport system. The alloys do not appear to be susceptible to hydrogen embrittlement. The extensive background of information, including experience gained from the

TABLE 2. Evaluation of Alloy Systems for Fussion Reactor Applications

	PATH A ALLOYS	PATH B ALLOYS	PATH C ALLOYS Ti	PATH C ALLOYS V,Nb
PRODUCTION AND FABRICATION (WELDING, ETC.)	○	◑	◑	●
PHYSICAL PROPERTIES (EXPANSION, CONDUCTIVITY)	◑	◑	◑	○
PROBABLE UPPER TEMPERATURE LIMIT (C)	550	650	600	(700), (800)
MECHANICAL PROPERTIES AT TEMPERATURES OF INTEREST (UNIRRADIATED)				
TENSILE	○	○	○	○ ○
FATIGUE	◑	○	?	? ?
CRACK GROWTH	◑	◑	?	? ?
CREEP-RUPTURE	○	○	○	○ ○
IRRADIATION INDUCED PROPERTY CHANGES				
SWELLING AND CREEP	●	●	?	?
MECHANICAL PROPERTY DEGRADATION	●	●	?	?
CORROSION, MASS TRANSPORT (Li SYSTEM)	○	◑	◑	◑
HYDROGEN SOLUBILITY (INVENTORY)	○	○	◑	○
(EMBRITTLEMENT)	○	◑	◑	◑
RADIOACTIVITY	◑	◑	○	○ ●

○ ACCEPTABLE, STATE-OF-ART; DEFINITE ADVANTAGE

◑ MARGINALLY ACCEPTABLE OR POTENTIAL PROBLEM AREA, TESTING AND DEVELOPMENT REQUIRED

● PROBABLY NOT ACCEPTABLE, SIGNIFICANT DEVELOPMENT EFFORT REQUIRED, DEFINITE DISADVANTAGE

? UNKNOWN

fast breeder reactor programs leads to identification of a prime candidate alloy having the composition given in Table 3. Optimization by control of microstructure and by minor alloying element additions will focus on improving mechanical properties which are degraded by transmutation-produced helium (tensile ductility, creep-rupture ductility and life, fatigue life, and crack growth) and control of irradiation induced swelling.

TABLE 3. Alloys Presently Under Investigation in the Fusion Reactor Alloy Development Program.

Path A:	Reference Alloy	
	20% CW 316	
	Prime Candidate Alloy	
	Fe-16 Ni-14 Cr-2 Mo-(Mn, Ti, Si, C)	
Path B:	Base Research Alloys	
	Fe-25 Ni-10 Cr-1 Mo	Plus Ti, Al C, Si Mn, B, Zr
	Fe-40 Ni-12 Cr-3 Mo	
	Fe-30 Ni-12 Cr-2 Nb	
	Fe-40 Ni-12 Cr-3 Nb	
	Fe-75 Ni-15 Cr-1 Nb	
Path C:	Scoping Alloys	
	Ti-6 Al-4 V	
	Ti-6 Al-2 Sn-4 Zr-2 Mo-0.25 Si	
Ti:	Ti-5 Al-6 Sn-2 Zr-1 Mo-0.25 Si	
	Ti-3 Al-8 V-6 Cr-4 Mo-4 Zr	
	V-20 Ti	
V:	V-15 Cr-5 Ti	
	Vanstar-7 V-9 Cr-3.3 Fe-1.3 Zr-0.054C	
Nb:	Nb-1 Zr	
	Nb-5 Mo-1 Zr	
Path D:	Scoping Alloys	
	2 LRO Alloys	

Path B — High Strength Fe-Ni-Cr Alloys

The γ' (coherent face centered cubic phase) and γ'' (body centered tetragonal phase) strengthened alloys offer higher strength and temperature capability than the basically solid solution strengthened austenitic alloys. Fast reactor irradiations, in which the He:dpa ratio is a factor of 40 below that in a fusion reactor, have resulted in significantly lower void swelling than in stainless steels. Irradiations in mixed spectrum reactors in which the helium generation is much higher than in a fast reactor have shown swelling levels comparable to stainless steels[9] indicating that improvements will be required. Irradiation embrittlement is a significant problem with this alloy system. Fabrication and welding problems related to the high strength, complex heat treatments, required cooling rates, etc., have been encountered for some precipitation hardened Fe-Ni-Cr alloys. In a lithium system corrosion of high nickel alloys will probably be unacceptable at 650°C. These alloys do, however, have good compatibility with helium and molten salts. There is evidence of reduced ductility and fracture toughness when exposed to high hydrogen pressures. Alloys of this system are in commercial production, and there is sufficient information on production, fabrication, strength properties and compatibility to demonstrate the potential application. A series of base research alloys (Table 3) has thus been selected. These alloys will be used to assess the effects of the fusion reactor environment on key properties and to identify those metallurgical variables which can be used to control behavior. The alloys were selected to represent the basic strengthening mechanisms which can be used: strengthened by γ' with either molybdenum or niobium as the austenite solid solution strengthener, strengthened by γ' and γ'' and a high nickel γ' strengthened alloy.

Path C — Reactive and Refractory Alloys

The reactive and refractory metal alloys are markedly different in character but are included together in Path C because their development schedules are judged to be similar.

Titanium alloys, based on unirradiated strength-to-weight ratio, fatigue, and creep-rupture properties, appear to be equal or superior to stainless steels for temperatures up to about 500°C. Limited data suggest that the alloys are compatible with lithium. The solubility of hydrogen in titanium alloys is relatively high and decreases with temperature. Thus, questions of tritium inventory in the structure and hydrogen embrittlement must be considered in the alloy development program. The most critical deficiency for the titanium alloy system is the lack of data on the response of these alloys to neutron irradiation. Four alloys have been selected for scoping studies. The alloys represent a broad range of compositions and microstructures and include ($\alpha + \beta$), α, and β alloys when classified according to the relative amounts of the α (HCP) and β (BCC) phases which they contain.

Refractory metal alloys may be the most promising candidates for achieving the long range goal of economic fusion power. Very limited results from fast reactor irradiation suggest that these alloys may be more resistant to irradiation swelling and embrittlement. Relatively good thermal conductivity and low thermal expansion will reduce thermal stress levels relative to Path A and B alloys. Industrial production capacity fabrication, welding, sensitivity to interstitial pickup and hydrogen embrittlement are major problem areas or questions which must be addressed. There is essentially no information on fatigue or crack growth properties. Three vanadium and two niobium alloys are presently included in the scoping phase of the program. The vanadium alloys have potential application to about 650 or 700°C and the niobium ternary alloy to about 800°C.

Path D — Innovative Materials and Concepts

Path D will address innovative material options. This path is included in the program in recognition that the conventional concepts of metallurgical development that are implicit in Paths A, B, and C may not provide optimum alloys for fusion reactor application. These alternative concepts will be examined to overcome the performance limits imposed by more conventional materials engineering. This path might include novel concepts such as composites, structural ceramics, unique material processes, ordered alloys, and other imaginative ideas that offer the possibility of reduction to engineering practice. It also will include ultra-low activation concepts. The development schedule will follow the steps in Path C after an initial evaluation of the concepts is completed.

MAJOR MILESTONES

A schedule of major (Level 1) milestones is given in Fig. 2. It allows an optimized alloy to be designed for each path by early in the 21st century. (OPT-A1 indicates the initial optimized alloy on Path A; the schedule allows two sequences of optimization on Path A and one on each of the others.) This is not to imply that all paths will be followed to completion; as results become available less promising approaches will be eliminated. This program presents the strategy for reaching an optimized alloy on each of the individual paths.

EXPERIMENTAL APPROACH

The goal of this program is the development of materials for use in the first wall and blanket structure of fusion reactors. There are numerous properties, environmental variables and synergisms that must be examined before the alloy which is finally developed is qualified for reactor design. Early in the program there are, however, too many alloys to investigate all properties and variables in detail. Thus, in each alloy system of interest the program must (1) identify the properties most critical to the successful use of that system; (2) determine if these properties are degraded by the fusion reactor chemical and/or irradiation environment; and (3) determine if those properties which are inadequate or degraded to unacceptable levels can be improved by the metallurgical variables of composition and/or structure. It is thus important that the testing program give results which can be used to estimate or predict properties under expected reactor operating conditions.

YEAR	PATH A	PATH B	PATH C	PATH D
1978	SELECT PATH A PRIME CANDIDATE ALLOYS (I–1)	SELECT PATH B BASE RESEARCH ALLOYS (I–6)	SELECT PATH C ALLOYS FOR SCOPING STUDIES (I–10)	ISSUE REQUEST FOR PROPOSALS FOR IDENTIFICATION OF PROMISING PATH D MATERIALS (I–15)
1979				
1980				SELECT PROMISING PATH D CONCEPTS FOR SCOPING STUDIES (I–16)
1981			SELECT PATH C BASE RESEARCH ALLOYS (I–11)	
1982				
1983	IDENTIFY OPT-A1 (I–2)	SELECT PATH B PRIME CANDIDATE ALLOYS (I–7)		SELECT PATH D BASE RESEARCH MATERIALS (I–17)
1984				
1985				
1986			SELECT PATH C PRIME CANDIDATE ALLOYS (I–12)	
1987				
1988				SELECT PATH D PRIME CANDIDATE MATERIALS (I–18)
1989				
1990				
1991	ESTABLISH ENGINEERING DATA BASE AND PERFORMANCE LIMITS FOR OPT-A1 (I–3)	IDENTIFY OPT-B (I–8)		
1992	IDENTIFY OPT-A2 (I–4)			
1993				
1994			IDENTIFY OPT-C (I–13)	
1995				
1996				IDENTIFY OPT-D (I–19)
1997				
1998				
1999	ESTABLISH ENGINEERING DATA BASE AND PERFORMANCE LIMITS FOR OPT-A2 (I–5)			
2000		ESTABLISH ENGINEERING DATA BASE AND PERFORMANCE LIMITS FOR OPT-B (I–9)		
2001				
2002			ESTABLISH ENGINEERING DATA BASE AND PERFORMANCE LIMITS FOR OPT-C (I–14)	
2003				
2004				ESTABLISH ENGINEERING DATA BASE AND PERFORMANCE LIMITS FOR OPT-D (I–20)

FIGURE 2. Comparison of Level 1 Milestones on Different Alloy Development Paths

Measurements of properties which are sensitive to the chemical environment must be done in the environment of interest. For example, fatigue and crack growth properties of stainless steels are about a factor of 10 lower in air than in a low oxygen partial pressure.(10) Use of results obtained from tests conducted in air to predict behavior of this alloy in a fusion environment will lead to a significant under-prediction of the true capability of the alloy. Testing in an appropriate chemical environment will be even more important for the refractory alloys based on vanadium and niobium.

The effects of irradiation on physical and mechanical properties will form one of the primary bases for alloy selection and optimization. The lack of a large volume irradiation test facility, with a neutron spectrum and flux characteristic of a fusion reactor, forces the program to rely on a variety of approaches which approximate the fusion reactor irradiation effects. The accuracy of the approximation will depend on (1) primary damage distribution, (2) damage energy deposition rates, and (3) transmutation product production rates. As reviewed by Wiffen and Stiegler(11) it has been established that the qualitative nature of the displacement damage produced by 14 MeV neutrons is identical to that produced by fission reactor neutron spectra. In low temperature-low fluence irradiations of pure metals small dislocation loops are produced in both spectra. In some experiments detailed agreement in cluster size distribution and total defect content has been obtained[12,13,14] while in other experiments small differences in size and distribution of clusters have been observed.[15,16,17] Comparisons of the response of a number of material properties to irradiation show that any given property varies similarly with fluence in the different neutron spectra. The effect of the neutron spectrum is to translate the property change along the fluence axis. Thus, at this time, it appears that a fission reactor spectra will adequately approximate the fusion reactor displacement damage. Benchmark experiments correlating the fusion and fission reactor spectra to establish the shift in fluence axis will be required. The primary problem with use of fission reactors is the inability to match the transmutation product production rates. The production of helium from (n,α) reactions is of particular importance because of the effects on swelling kinetics and mechanical properties. Fortunately in mixed spectrum fission reactors, where both the fast and thermal fluxes exceed 3×10^{18} n/m^{2}s, both helium and dpa production rates equivalent to those of a fusion reactor at 1 MW/m^2 can be achieved in alloys containing nickel. Displacements are produced by the fast neutrons and helium is produced by the two step reaction

$$^{58}\mathrm{Ni} + n_{(\mathrm{th})} \xrightarrow{\sim 4\mathrm{b}} {}^{59}\mathrm{Ni}$$

$$^{59}\mathrm{Ni} + n_{(\mathrm{th})} \xrightarrow{12\mathrm{b}} {}^{56}\mathrm{Fe} + \alpha \ .$$

However, as shown in Fig. 3(a), irradiation in a mixed-spectrum reactor where the thermal-to-fast neutron ratio is approximately constant during the entire irradiation, the same He/dpa as that in an MFR first wall is achieved at only one time during irradiation. The helium concentration increases approximately as $(\phi_{th}t)^{1.5}$ and the dpa increases linearly with $(\phi_f t)$. Hence, the time when the He/dpa in a fission reactor exceeds that in an MFR is a function of the ratio of thermal-to-fast neutron flux. To utilize fission reactors for MFR first-wall development of materials that contain nickel the He/dpa during the irradiation must closely match that anticipated in an MFR. This can be achieved by tailoring the spectrum through adjustments in the amount of fuel and moderator around the experiment. In the example shown in Fig. 3(b), extremely simple spectrum tailoring of an experiment in the Oak Ridge Research Reactor was used. No attempt was made to optimize the tailoring of the spectrum to produce a He/dpa closer to the MFR value. How close the He/dpa must be controlled to simulate the effects of the MFR environment is not yet known. Thus for Path A and B alloys the alloy development program will utilize mixed-spectrum fission reactors with spectral tailoring to approximate the helium and dpa production of a fusion reactor. The experimental volume available is sufficient to carry out the extensive testing required in the alloy development program. Fast reactors (EBR-II and FFTF) will be used to examine response at high dpa levels (with He/dpa much less than in a fusion reactor) and high flux mixed spectrum reactors, such as the HFIR, will be used to achieve helium levels equivalent to several years fusion reactor exposure (with He/dpa levels much greater than a fusion reactor). The possibility exists to do sequential irradiations in reactors such as HFIR and FFTF to attain goal dpa and helium levels in a stepwise manner.

Initial reactor irradiations will emphasize postirradiation testing and examination. The capability for performing in-reactor experiments in which temperature, stress, and strain are measured and varied in a controlled way will be developed as rapidly as possible. Because of the large number of materials and conditions to be considered a major early effort will be devoted to miniaturization of test specimens. Attention will be concentrated on mechanical properties measurements

Microstructural observations will be used as an interpretive tool rather than a primary experimental objective.

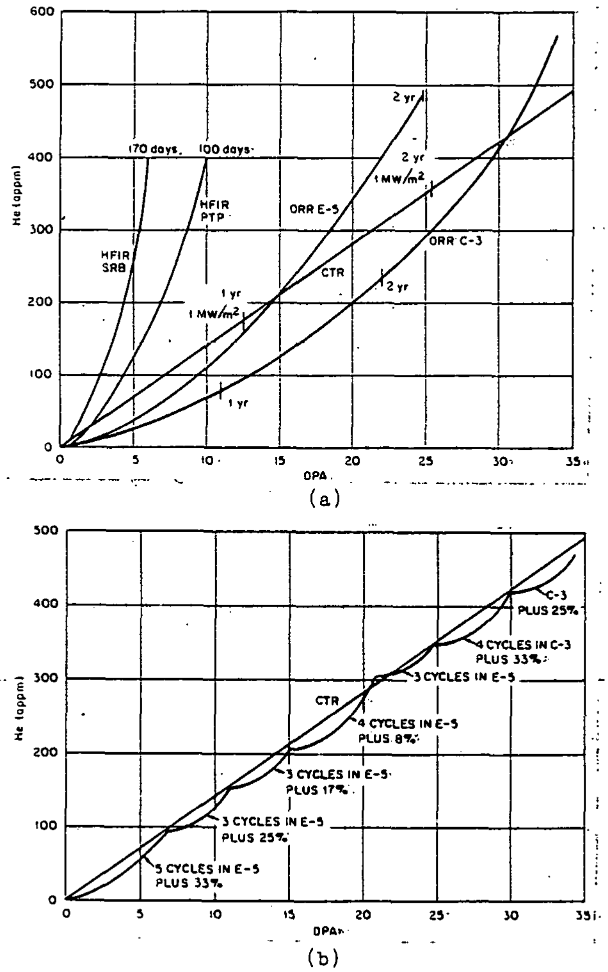

FIGURE 3. Helium Concentration as a Function of Displacements per Atom in Type 316 Stainless Steel for an MFR First Wall and for Irradiation in (a) Two HFIR and Two ORR Positions, (b) Two Positions of ORR with Spectrum Tailoring. Percentage values are for the increases in the thermal neutron flux produced by varying the moderation between the reactor fuel and the test samples.

The situation for alloys on Paths C and D is less satisfactory. Fission reactors again will provide the main test vehicle, but because of their spectral characteristics transmutation rates will be much below those of a fusion reactor spectrum. This limitation will be overcome to some extent by injecting helium and hydrogen into the specimens prior to neutron irradiation or by doping the specimens with isotopes that have acceptable cross sections for (n,p) and (n,α) reactions. This approach will probably not be useful for generating an engineering data base, but it will provide the guidance for early alloy development decisions. If such alloys are identified for commercial fusion reactors, a large volume, high flux fusion test facility will be required. Alloy development on these paths will also be aided by ion bombardment studies in which heavy ions produce displacements while light ions (H and He) are simultaneously injected at appropriate rates. Because of our relatively meager knowledge of the properties of these materials in the absence of irradiation, much of the guidance in the development of alloys will come from studies of the physical and mechanical metallurgy of unirradiated materials. This will greatly reduce the number of materials to be subjected to irradiation testing.

The Fusion Materials Irradiation Test Facility (FMIT) and the Damage Analysis and Fundamental Studies Programs are important parts of the overall alloy development strategy. In the FMIT the experimental volume with a high neutron flux ($>10^{18}$ n/m^{2}s) is small (∿0.5 liter). Alloy development must thus rely on fission reactors because of their large experimental volume. Experiments in FMIT will provide the link between the fission reactor data and the effects of the fusion reactor neutron spectrum.

SUMMARY

The objective of the Alloy Development for Irradiation Performance Program is the development of structural materials for use in the first wall and blanket region of fusion reactors. The goal of the program is a material that will survive an exposure of 40 MWyr/m^2 at a temperature which will allow use of a liquid metal-H_2O heat transport system. Although the ultimate aim of the program is development of materials for commercial reactors by the end of this century, activities are organized to provide materials data for the relatively low performance interim machines that will precede commercial reactors.

Most, if not all, materials properties which are critical in fusion reactor design are sensitive to the alloy composition and/or microstructure even in the fusion reactor environment. This means that in any alloy system improvements in performance are possible through control of these quantities. The first step is to identify the most promising alloy systems and then to optimize or tailor their performance for fusion reactor applications.

The range of problems, irradiation conditions and possible solutions in terms of composition and microstructure is so vast that a trial and error or Edisonian approach to alloy development is unlikely to succeed. A disciplined approach in which the effects of important material and irradiation variables are understood, at least in qualitative terms, and applied to the design of alloys is essential to meeting the goals of this program.

Initial studies suggest that the austenitic alloys, higher strength Fe-Ni-Cr alloys and the refractory or reactive metal alloys all offer attractive properties for fusion reactor applications. The Alloy Development Program will follow a parallel path approach including; austenitic alloys, higher strength Fe-Ni-Cr alloys, refractory/reactive alloys, and innovative concepts. The latter path is included in the program in recognition of the possibility that the conventional concepts of alloy development may not be adequate for all fusion reactor applications. This path will include novel approaches that offer the possibility of reduction to engineering practice and improved performance of the fusion reactor system.

Although this plan describes the steps for producing an optimized alloy on each of four parallel paths, this is not intended to imply that all paths will be followed to completion. As the program develops, the less promising approaches will be eliminated, and attention will be concentrated along the most productive lines. During the first five or six years the program is designed to fill two needs: first, an optimized austenitic alloy will be developed and a data base suitable for design of interim machines will be produced; and second, scoping work will be completed to identify promising alloy development approaches on the other paths. At that time the reactor design concepts should become better defined as a result of plasma physics experiments now under construction, and the alloy development effort can then be focused to meet those needs.

The exposure goal of 40 $MWyr/m^2$ makes irradiation effects the dominant element in the program. There is no way to estimate the properties of materials irradiated to such high levels from the properties of unirradiated materials. Final decisions on alloy choices will be based on extensive measurement of irradiated materials including the evaluation of properties during irradiation. The program will not be directed exclusively at irradiation performance, however, and much of the early identification of promising alloy systems will be based on measurement of properties outside the irradiation environment.

The ideal test environment has a fusion neutron spectrum, a flux high enough to allow accelerated testing and an experimental volume on the order of several liters. Such a source is not currently available or authorized, but the plan assumes that one will become available around 1990. During the intervening time the program must rely on less-than-ideal sources (principally fission reactors), simulation techniques, and, of necessity, a strong understanding of the physical processes that occur during irradiation. The Damage Analysis and Fundamental Studies Task Group and the Fusion Materials Irradiation Test Facility are essential elements of this part of the strategy, for they will allow us to relate fission reactor behavior to the fusion reactor environment.

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