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ABSTRACT

Fusion reactor irradiation response in alloys containing nickel can be simulated in thermal-spectrum fission reactors, where displacement damage is produced by the high-energy neutrons and helium is produced by the capture of two thermal neutrons in the reactions: $^{58}Ni + n \rightarrow {}^{59}Ni + \gamma$; $^{59}Ni + n \rightarrow {}^{56}Fe + \alpha$. Examination of type 316 stainless steel specimens irradiated in HFIR has shown that swelling due to cavity formation and degradation of mechanical properties are more severe than can be predicted from fast reactor irradiations, where the helium contents produced are far too low to simulate fusion reactor service. Swelling values are greater and the temperature dependence of swelling is different than in the fast reactor case. The property change most restrictive for fusion reactor performance is the low values of ductility that result from irradiation. These results imply limitations on the operating conditions and useful lifetimes of stainless steel first-wall and high flux region structural components of fusion reactors.

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1. INTRODUCTION

When the remaining questions of physics and engineering are solved, and electrical power from magnetically confined fusion reactors becomes a reality, the reliability and cost of this power will be heavily influenced by the properties of the structural materials. One of the possible classes of materials that could be used in these structures is the austenitic stainless steels. Type 316 stainless steel is representative of this alloy class and is an alloy with a large body of engineering properties data. Examination of these data, and of the operating requirements of fusion power systems, define the first-level of materials-imposed limitations on these reactors. The changes in properties under conditions that simulate the fusion reactor environment produce a second-level of materials-imposed system limitations. These results also point the direction for additional property evaluations.

Materials-imposed limitations on fusion reactor component lifetime may occur through restrictions on design stresses, operating temperature, system power density, or number of operating cycles.

Evaluation of the response of materials to irradiation in a fusion reactor environment can best be determined after exposure of the material in a fusion reactor. Since there is no fusion reactor available, nor is a fusion neutron spectrum with adequate flux available, irradiation effects must be evaluated in simulation experiments. Although these partial simulations are less-than-ideal, there is no alternative to their use.

The most damaging component of the fusion reactor environment is the flux of high energy neutrons on structural components near the plasma chamber. The irradiation of nickel-containing alloys in the High Flux Isotope Reactor (HFIR) provides a good simulation of several components of the fusion reactor service. Examination of property changes measured in type 316 stainless steel specimens irradiated in HFIR are used in this paper to forecast the response of this alloy to fusion reactor service.

2. THE FUSION REACTOR ENVIRONMENT

The generation of electrical power in a magnetically confined fusion reactor (MFR) requires a vacuum containment chamber operating at a high enough temperature for the operation of a heat engine. The containment must be absolute; no leak can be tolerated that would admit coolant or breeding fluids into the plasma chamber. Once the reactor has operated for any significant period, the structure will be highly radioactive, and repair of leaks will become very difficult. Structural materials that will have a long service life thus have a major impact on the system economics. Several analyses of the effects of structural lifetime suggest that a minimum acceptable life is two to four years, and that significant economic gains result from achieving lifetimes of five to ten years or greater.[1,2] These required service lifetimes cannot be more accurately defined, since too many assumptions are required. The system temperature requirements are also not precisely defined. The reactor coolant outlet temperature should probably not be lower than about 300°C. An upper limit on usable coolant outlet temperature is likely 600 to 650°C. Operating temperatures in a fusion reactor will allow a gradient of approximately 100°C between coolant inlet and coolant outlet. The temperatures of reactor structural components will then be determined by the gradients imposed by energy transfer to the coolant.

The power level of a Tokamak reactor is specified in terms of the neutron wall loading, that is the energy carried through the first wall by the current of source neutrons from the d-t fusion reaction. Neutrons carry 80% of the energy produced in the fusion reaction. Values specified in conceptual reactor designs have ranged from 0.1 to 10.0 MW/m^2, and current design calls for neutron wall loadings of 1 to 5 MW/m^2. The value of the wall loading does not set the system temperature, but it does have a strong influence on the temperature gradients within the system during the plasma burn and the temperature variation between "plasma burn" and "plasma off" portions of the operating cycle. An additional power load on the system is delivered in the non-penetrating radiation incident on the first material wall facing the plasma. This amounts to about 20% of the total energy produced in the plasma reaction and can reach the wall carried by energetic ions (helium or hydrogen isotopes) and x-rays at various energies. All this energy is deposited at or very near the first surface. It appears probable that the first structural wall will be coated or lined in some way to minimize effects of this non-penetrating radiation. For this discussion of a stainless steel structural first wall, this loading is important only as a major contributor to temperature gradients.

Neutrons produced by the plasma make frequent collisions with the lattice atoms. Energetic collisions result in displacement of lattice atoms or, at high energies, trigger cascades of multiple atom displacements. For purposes of quantifying the amount of irradiation, the number of times each atom is displaced is calculated, and the displacements per atom, dpa, are reported as an energy-weighted measure of the integrated neutron flux. The neutron-lattice interaction can also be non-elastic, with the collision triggering any one of a large number of possible nuclear reactions. Gas production reactions are the most common, yielding hydrogen or helium reaction products, and a measure of the total neutron exposure may also include the rate of production of these gases. The number of displaced atoms and gaseous reaction products in type 316 stainless steel for one year of operation of a fusion reactor are given in Table 1. Also given are the same data calculated for several fission reactors. Comparison of these rates for fusion and fission reactors shows the advantages and limitations of the use of these reactors to simulate anticipated conditions of fusion service.

3. SIMULATION OF THE FUSION REACTOR ENVIRONMENT

With no fusion reactors available for testing the response of candidate structural materials to this unique environment, this testing requires facilities that simulate as many as possible of the key features of the fusion power system. In this paper the interest is in the irradiation environment, and other components of fusion service such as the chemical environment and mechanical loading are not treated.

Table 1. Defects Produced in Type 316 Stainless Steel for One Year Operation of Various Facilities[b]

Reactor[a]	Neutron Fluence	Displacements	Helium	Hydrogen
	$\times 10^{26}$ n/m^2, >0.1 MeV	(dpa)	(appm)	(appm)
Fusion Reactor (1 MW/m^2)	0.80	11.5	144	530
EBR-II	6.9	37	∿20	300
HFIR	4.4	35	1900[c]	425
ORR	1.3	11	80 to 160	135

(a) A high flux position has been chosen for each reactor, and a duty faction of 1.0 has been assumed.

(b) Results from ORNL calculations.

(c) Measured value.

The experience of 35 years study of the effects of irradiation on solids has provided a wealth of guidance on the approach to be used in investigating a new environment such as will be encountered in fusion reactors. This experience has shown that we must perform irradiations at the temperature of interest; our understanding of irradiation response is not adequate for temperature extrapolation. We have also learned that it is only possible to make limited extrapolations of fluence, thus our experiments must come very close to matching the anticipated service lifetime if the definition of end-of-life properties is a prime goal. It is also clear that not all properties can be measured after the termination of the irradiation. Some properties of interest are dependent on the point defect concentrations during irradiation and these properties can only be measured on specimens while they are in the defect-producing flux. The best example of a property that must be measured during irradiation is creep, where in-reactor and out-of-reactor behavior show little or no similarity.

The calculational results in Table 1 show the match in some parameters for anticipated fusion and currently available fission reactors. Matching the rate of displacement damage production is not a problem; the Oak Ridge Research Reactor (ORR) will match fusion reactors for a modest power level, and the Experimental Breeder Reactor (EBR-II) and HFIR match for power densities more in line with current design thinking, 3 to 4 MW/m^2. The fusion reactor rate of hydrogen production is not matched in any fission reactor; the ratio of hydrogen-to-dpa production rates is low in fission reactors by a factor of four to six. This is believed to be of little concern for stainless steels at the proposed temperature of application. Although definitive experiments are still required, it seems likely that in the range 250 to 550°C the hydrogen will readily diffuse out of the structures to be gathered in the D and T fuel handling system of the reactor. Simulation of the helium that results from single-step (n,α) reactions between the <u>high</u> energy neutrons of a fusion reactor and all the alloy constituents of stainless steels is a matter of greater concern. Helium has been shown to have deleterious effects on materials' properties, and the extent of these effects must be known for fusion applications. Helium production rates of interest can be achieved in some fission reactors through a unique reaction sequence between thermal energy neutrons and the major isotope of nickel:

$$^{58}Ni + n \longrightarrow {}^{59}Ni \quad \ldots\ldots\ldots\ldots \quad (1)$$

$$^{59}Ni + n \longrightarrow {}^{56}Fe + {}^{4}He \quad \ldots\ldots\ldots \quad (2)$$

Since this reaction sequence involves thermal energy neutrons, it is <u>not</u> of importance in fusion reactors or fast-spectrum fission reactors but is important in mixed spectrum fission reactors with a large fraction of the neutrons at thermal energies. Production rates are shown in Table 1.

The final component of the fusion reactor environment that must be considered in experiments designed to simulate that environment is the duty cycle. Several fusion reactor concepts involve pulsed operation. In particular, the Tokamak reactors will operate in a long-pulse mode, with pulses of hundreds to thousands of seconds interrupted by "recylce pulses" of a few to tens of seconds. The neutron bombardment is "on" during the long or burn fraction of the cycle and "off" during the recycle portion. In general this cycle cannot be matched in fission reactors, although the resulting temperature cycle can probably be simulated during fission reactor experiments.

The balance of this paper will examine the effects of irradiation in HFIR, which provides a useful simulation of the temperature, displacement production, and helium production anticipated for fusion reactors, on the properties of type 316 stainless steel. Details of the experimental methods and a more complete coverage of the experiment results are given in references 3, 4, and 5.

4. MICROSTRUCTURAL CHANGES PRODUCED BY IRRADIATION

The high mobility of individual vacancies and interstitials at fusion reactor operating temperatures results in most point defects being annihilated soon after they are created. A small fraction of these defects are retained by forming vacancy or interstitial clusters. These relatively stable defect configurations influence the material properties. Mobile interstitial atoms precipitate as dislocation loops. Vacancies can precipitate in two morphologies, either dislocation loops or three-dimensional cavities. Two limiting classes of cavities can be defined for evaluation of fusion reactor damage. Cavities that are essentially empty, formed by the precipitation of vacancies alone, are referred to as voids. Cavities that form by precipitation of both vacancies and insoluble gases (e.g., helium) are called bubbles and can exist in equilibrium with an internal pressure, P, given by

$$P = \frac{2\gamma}{r} \tag{3}$$

where γ = specific surface tension of the metal and r = bubble radius. Intermediate cavity states are possible, with cavities neither pure void nor pure bubble, and these are expected for fusion reactor service.

Precipitation of second phases, and the accompanying microsegregation of alloy constituents, can be enhanced by neutron irradiation. These local compositional changes can in turn affect other components of the irradiation response as beneficial alloying elements are removed from solution. Precipitation in type 316 stainless steel irradiated in the HFIR reactor, where both high dpa level and high helium contents were achieved, is discussed in detail by Maziasz.[6]

Figure 1 shows microstructures of type 316 stainless steel irradiated in the HFIR reactor. The cavities have been shown to be near-equilibrium helium bubbles.[5] The dislocation structures are composed of network dislocations and perfect dislocation loops. Material irradiated to the conditions listed in Fig. 1 does not contain Frank loops. Material irradiated to lower fluences in HFIR does contain Frank loops. The trend with increasing irradiation temperature is for cavity concentrations to decrease and size to increase. Dislocation structures generally coarsen with increasing temperature, and precipitate concentrations decrease and sizes increase.

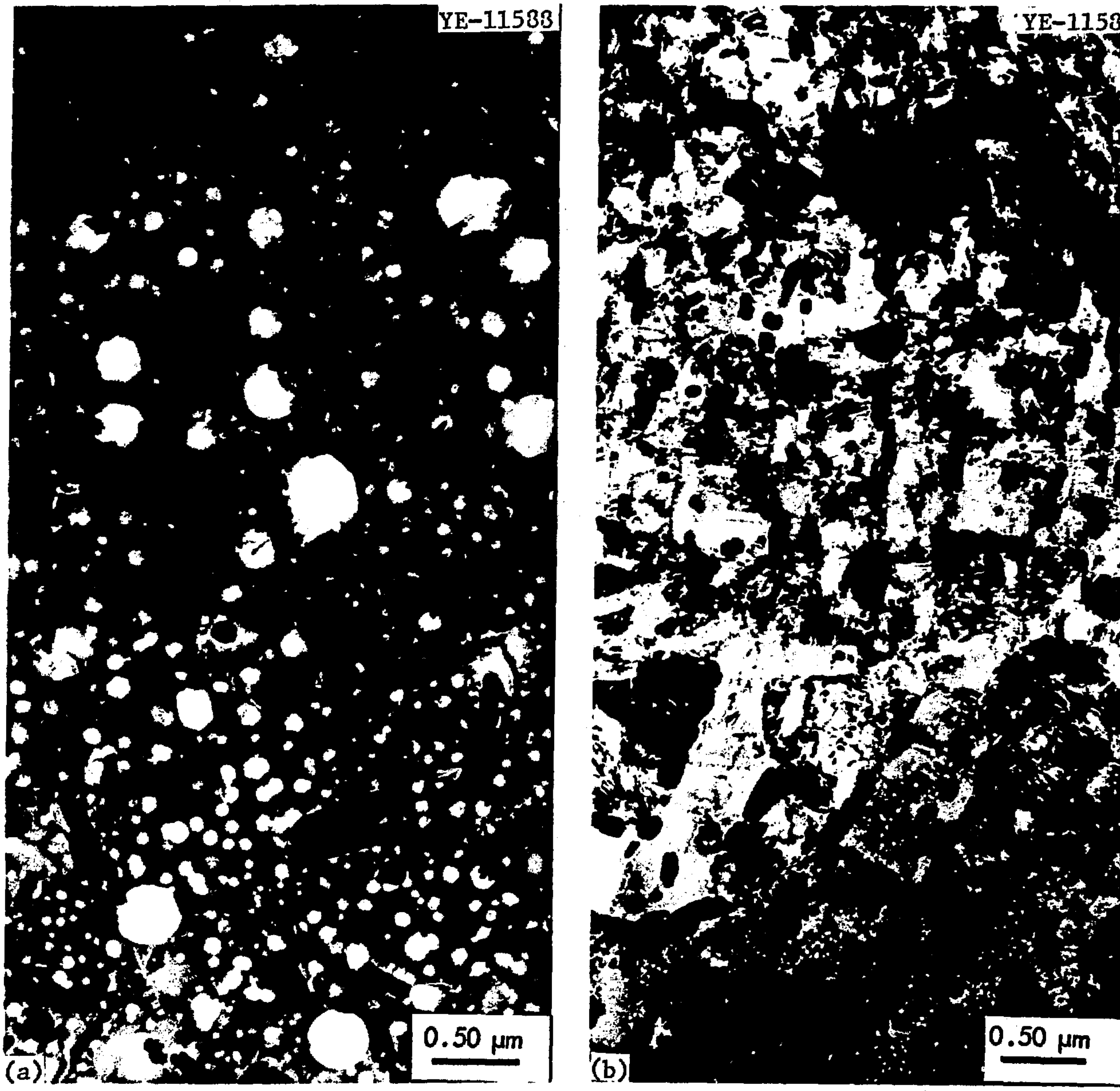

Fig. 1. Type 316 Stainless Steel Irradiated in the HFIR at 550°C to a Fluence of ∿6 × 10^{26} n/m^2 (>0.1 MeV), Producing 42 dpa and 3000 appm He. (a) Material solution annealed 1 hr at 1050°C before irradiation. Cavity volume fraction is ∿8%. The precipitates shown are $M_{23}C_6$ and M_6C (larger precipitates), and Laves (lath-shaped precipitate). (b) Material 20% cold-worked prior to irradiation. Cavity volume fraction ∿1.5%. Cold-work is partially recovered to form cell structure. Note that at this temperature the grain boundary cavities are larger than the matrix cavities. Precipitate structures are treated in detail by Maziasz (Ref. 6).

5. CHANGES IN ENGINEERING PROPERTIES

The changes in microstructure that result from irradiation change the properties of the material. This becomes important when the property changes are great enough to influence the operating conditions and useful lifetime of reactor structural components. The engineering properties affected by irradiation that will be discussed here are:

swelling,
tensile properties,
creep rupture life,
fatigue and crack growth, and
in situ properties.

After changes in these properties have been discussed, the implication on the operation of fusion reactors will be considered.

(a) Swelling

Swelling of stainless steel components in fusion reactors will result from the simultaneous condensation of vacancies and insoluble gases [helium from (n,α) reactions] to form cavities that contain gas at or near the equilibrium pressure. General considerations that relate to swelling are summarized in Table 2. The generally accepted description of the void swelling processes in metals, with little gas production during irradiation, is as follows. While self-interstitials and vacancies are produced in equal number by neutron collision with the lattice, the preferential attraction of dislocations for self-interstitials provides a biased sink that results in an excess of free vacancies in the

Table 2. Swelling in Irradiated Stainless Steel

<u>Cause</u>:

Vacancies and gas atoms precipitate in cavities
Interstitials precipitate in loops or on dislocations
Net volume increase results

<u>Critical Parameters</u>:

Fluence (displacement damage and gas production)
Temperature of irradiation
Composition
Metallurgical state

<u>Modifying Parameters</u>:

Damage rate
Stress state
dpa/He ratio
Duty cycle?

<u>Characterization of Swelling</u>:

Overall dimensional changes
Dimensional changes between ion-bombarded and unbombarded regions of a sample
Density by immersion
Microstructure by TEM

lattice. These vacancies then provide the driving force for nucleation and growth of cavities and the resultant swelling. The role of transmutation-produced helium in this process is not clearly understood, but it is likely that the helium can enhance swelling by its effect on both the nucleation and growth of cavities.

Measured cavity parameters and amount of swelling for samples of 20% cold-worked type 316 stainless steel are given in Table 3. These samples were irradiated in HFIR over a range of temperatures, to neutron fluences producing dpa levels equivalent to 3.7 to 5.3 $MWyr/m^2$ and helium levels equivalent to 21 to 29 $MWyr/m^2$ of service in the first wall zone of a fusion reactor. Detailed analysis of these swelling values and comparison with swelling information generated under irradiation that produces similar dpa levels but only 5 to 50 appm He lead to the following conclusions:[3,5]

i) Greater swelling occurs with high helium contents compared to irradiations producing helium levels well below those anticipated in fusion application.

ii) Swelling with high helium contents is relatively insensitive to irradiation temperature in the range 380 to 600°C. Greater swelling occurs at higher irradiation temperatures.

iii) At all temperatures, the effect of helium on cavity formation is mainly through an increase in cavity concentration. This is assumed to be through its effect on the cavity nucleation stage. At irradiation temperatures above 600°C, size is also significantly increased by the helium.

Table 3. Cavity Parameters for 20% Cold-Worked Type 316 Stainless Steel Irradiated in HFIR.[a]

Irradiation Temperature (°C)	Neutron Fluence ($\times 10^{26}$ n/m^2, >0.1 MeV)	dpa	Helium (appm)	Cavity Parameters: Diam (nm)	Cavity Parameters: Concentration ($\times 10^{21}$ m^{-3})	Cavity Parameters: Volume Fraction (%)	Cavity Nature[b]
380	7.1	49	3320	9.5	18	2.2	0.87
455	7.7	54	3660	17	6.6	2.0	0.77
550	6.2	42	2990	21	2.4	1.4	1.14
600	8.7	60	4070	65	0.33	5.0	0.82
680	8.7	61	4140	110	0.063	16.8	0.70

[a]Data from reference 5.

[b]This is the ratio of the measured helium content of the sample to the calculated amount of helium required to fully stabilize the observed cavity population as equilibrium gas bubbles.

iv. Cold work remains effective in reducing the level of swelling for these high helium concentrations for temperatures up to 600°C. At higher temperatures recovery and then recrystallization of the cold-worked structure results in swelling behavior similar to the solution annealed material.[3,5]

v. The observed cavities contain helium at or near the equilibrium pressure for the irradiation temperature.

(b) Tensile Properties

Irradiation of stainless steels under conditions simulating fusion reactor service results in major changes in the postirradiation tensile properties.[4] Data from the same sample set used to illustrate swelling effects are given in Fig. 2. In this experiment samples of 20% cold-worked type 316 stainless steel were irradiated at temperatures between 350 and 700°C to neutron fluences producing about 50 dpa and 4000 appm helium. The results in Fig. 2 show that this irradiation has significantly reduced both the yield and ultimate tensile stress over the range of temperatures investigated. This weakening results from recovery of the cold-worked structure and precipitation that occurs during irradiation. (At temperatures above 600°C intergranular separation leads to failure at low stresses.) While this decrease in tensile strength is important, and must of course be taken account of in reactor design, of even greater importance is the loss of ductility. Figure 2 shows that for the lowest irradiation and test temperature, 350°C, the ductility of the irradiated sample was only slightly less than that of the control. At increasing temperatures the tensile elongation continually

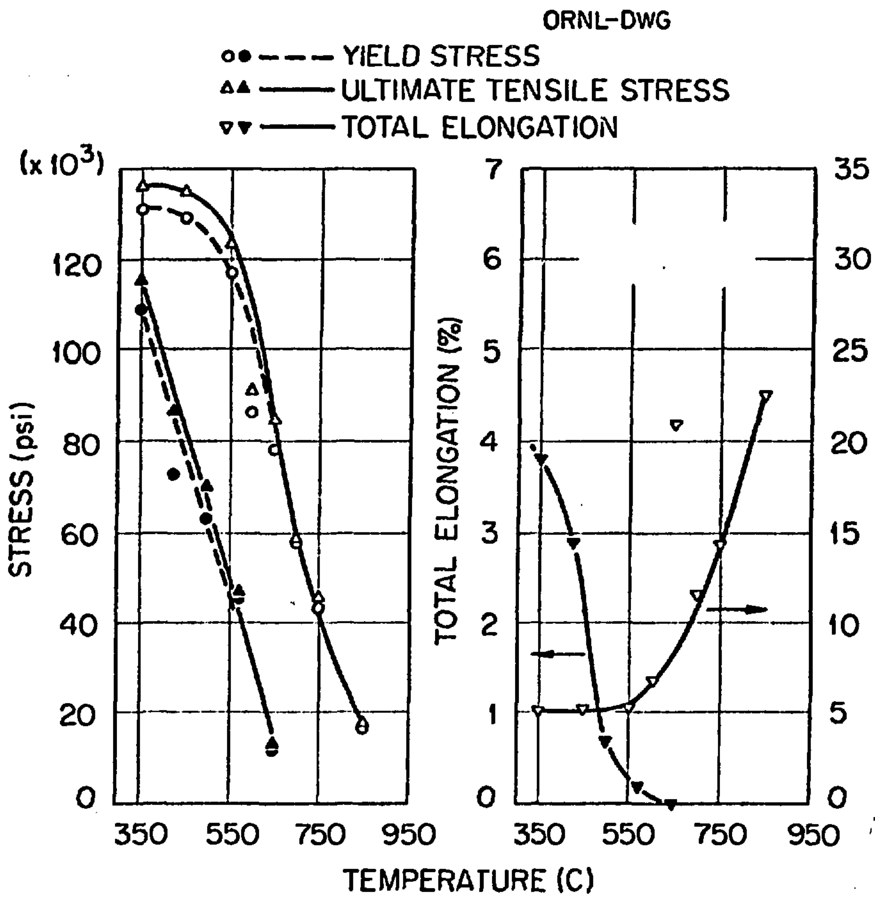

Fig. 2. The Tensile Properties of 20% Cold-Worked Type 316 Stainless Steel as a Function of Test Temperature. Open symbols are for unirradiated material, and closed symbols for samples irradiated at a temperature near the test temperature. The strain rate was 0.0028 min^{-1}. The HFIR irradiation produced about 50 dpa and 4000 appm helium in the samples. From [4].

decreases, dropping to zero at 650°C. These data clearly set an upper temperature limit on the use of this alloy, although the minimum allowable ductility has not yet been defined by reactor designs. The dependence of tensile ductility on reactor operating time must also be known to predict reactor component lifetimes. The type of data used for this purpose is shown in Fig. 3 where tensile tests results at 350 and 575°C are shown as a function of neutron fluence. The irradiation temperatures were near the test temperatures. These data show that the ductility is rapidly reduced under these conditions and that helium accumulations of less than 100 appm (less than 0.5 $MWyr/m^2$ of MFR operation) can severely impair the material properties.

The fracture at the higher temperatures is dominated by the helium, with low ductilities resulting from grain boundary separations. Microstructural evaluation showed large cavities on most grain boundaries, and fracture likely occurs by some process involving the growth and linking of these cavities.

c) Creep Rupture Life

Results of uniaxial creep-rupture tests at 550°C and the high stress level of 310 MPa (45,000 psi) on samples irradiated to different damage levels at 545–605°C are shown in Fig. 4. The rupture life showed little change for the lowest fluence irradiation, and then decreased continuously with increasing damage level. At the highest damage level the rupture life was 2×10^{-5} of the unirradiated value. In contrast, irradiation of this same heat of steel in EBR-II to 7.5–15 dpa and 4–8 appm helium produced a slight increase above the unirradiated 550°C rupture life.[7] Thus, the marked reduction in rupture life appears to be associated primarily with

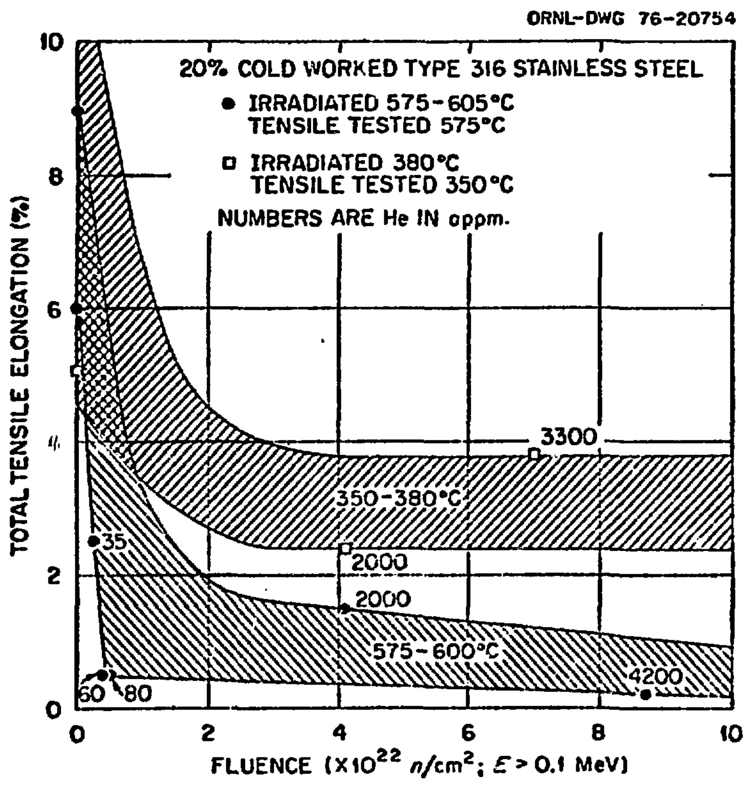

Fig. 3. Fluence Dependence of Total Elongation in HFIR Irradiated Cold Worked Type 316 Stainless Steel.

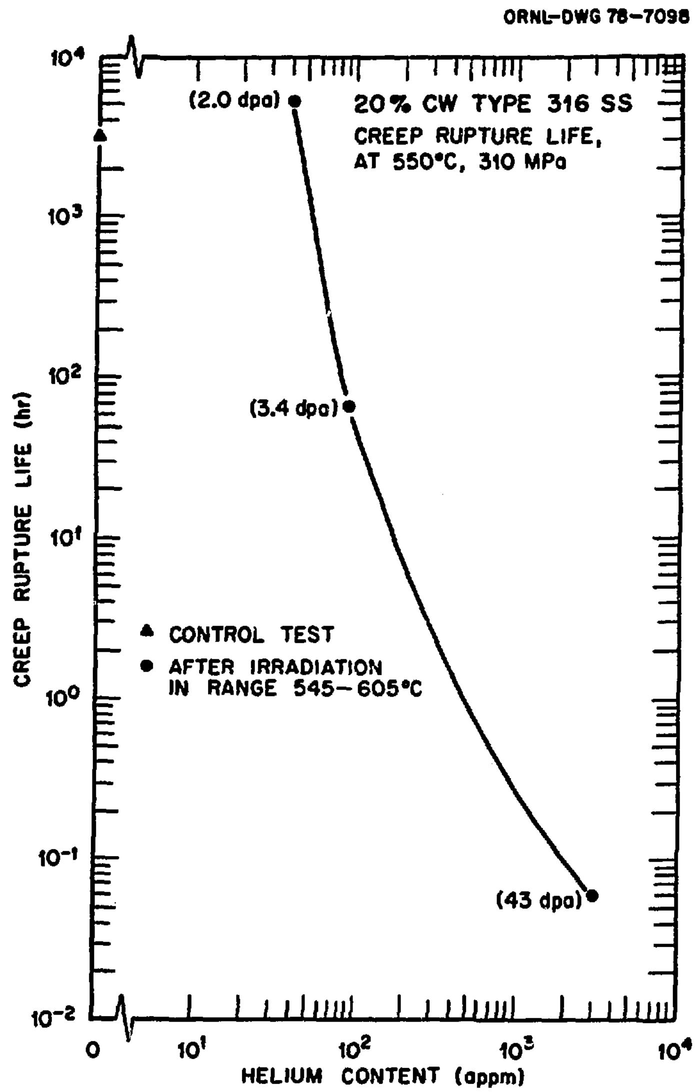

Fig. 4. Creep Rupture Life of 20% Cold-Worked Type 316 Stainless Steel Irradiated to Produce High Helium Concentrations. Stress level 310 MPa (45,000 psi).

the presence of the large amounts of helium. There are insufficient data to determine the extent of the observed property loss as a function of temperature and stress. Models of helium embrittlement predict a decreasing effect with decreasing stress.[8] This is confirmed to some extent by experimental results.[9]

(d) Fatigue and Crack Growth

Analysis of conceptual designs for Tokamak fusion reactor power systems[10] has shown that fatigue and fatigue crack growth may be lifetime-limiting properties. A problem with attempting detailed analysis of the response of structural components to cyclic loading is the lack of relevant test data. In the case of stainless steel most of the data, including that used in making lifetime predictions, are for an air environment. At low oxygen activity, typical of that in a fusion reactor environment (vacuum or lithium), the crack growth rates are a factor of 15 lower than in air. Furthermore, no fatigue tests have been conducted on stainless steel irradiated to high dpa levels and high helium contents.

Michel has surveyed the available data on the effects of irradiation on fatigue crack growth in stainless steels.[11] For irradiation temperatures in the range 400 to 500°C, and tests at 427°C, there was little or no effect of irradiation on the crack growth rate. For the same range of irradiation temperatures, tests at 593°C showed that the effect of irradiation was to increase the crack propagation rate. The tensile and creep-rupture test results discussed earlier showed that tests in the range 550 to 600°C exhibited severe ductility reductions, while lower temperatures had much less severe effects. There is some suggestion that the reduction in crack growth resistance may result from the enhanced intergranular fracture that accompanies high temperature irradiations. If in fact this is the controlling process, then the high helium contents characteristic of fusion reactor conditions would enhance crack growth rates for temperatures above ∿550°C. It is likely that this "response temperature limit" will be related to the helium content. The large amounts of helium will reduce the temperature at which the transition from transgranular to intergranular fracture occurs.

(e) *In-Situ* Properties

Some properties require measurement while the material is under irradiation. Of the properties requiring *in situ* measurement, the best known example is irradiation creep.(12) A point form treatment of irradiation creep is given in Table 4.

In situ measurements are required when changes in the property of interest are a result of the flux of point defects produced by the irradiation, rather than of the microstructure that has resulted from the precipitation of point defects. In the case of irradiation creep, the deformation rate of a stressed material is controlled by the biased flow of point defects to dislocations. Another property that may require *in situ* measurement is low-cycle fatigue.

Irradiation creep data on type 316 stainless steel have been generated in several fast spectrum fission reactors. These results show:(12)

Table 4. Irradiation Creep

Cause:

Biased flow of point defects control deformation
Deformation rates exceed out-of-reactor rates

Critical Parameters:

Flux — atomic displacement rate

Modifying Parameters:

Fluence
Stress
Temperature
Composition and microstructure
[High helium generation rate will modify the microstructural evolution, and affect irradiation creep rate.]

Measure:

Temperature
Stress
Strain (with high precision)

i. creep rate increases with increased displacement level;

ii. creep rate increases with increasing temperature but is much less temperature dependent than is thermal creep. Irradiation creep probably shows similar temperature dependence to that of swelling.

iii. Irradiation creep is nearly proportional to stress; and

iv. relatively high total strain can be achieved without specimen failure.

These data show that, for 20% cold-worked 316 stainless steel, irradiation creep is the dominant creep mechanism for temperatures up to about 600°C at the displacement rates found in EBR-II. Since irradiation creep rate is primarily dependent on displacement rate, these trends should be expected to prevail in fusion reactors.

Measurements are not available for irradiation under conditions which better simulate fusion reactor service.

6. IMPLICATIONS OF MATERIAL PROPERTY CHANGES ON FUSION REACTOR PERFORMANCE

Changes in the properties of reactor structural components will control the reliability and impact the economics of the power system. In analyzing the effect of 316 stainless steel properties on a fusion power system performance, the state of development of this alloy is an advantage. This alloy is backed by an extensive history of production and application, including use in fission reactors. As a result, there will be an extensive data base on unirradiated material that can be used in reactor design. As the component service time in the reactor builds up, the increasing neutron fluence will result in increasing property changes. Limitations on irradiation space for testing programs also result in a limited body of data at high fluence, and thus the accuracy of design predictions of service performance will be a decreasing function of service time.

Type 316 stainless steel is a strong, ductile material in the unirradiated state. During initial service in a reactor, in shake-down and early power production operation, it will be relatively forgiving of unpredicted loading conditions. The result is that the structure should deform to minimize stress during the reactor power cycle operation long before the service irradiation has significantly affected properties.

In broad, general terms the irradiation effects data reviewed in this paper suggest limitations on temperature, stress, strain, and lifetime of type 316 stainless steel reactor components. Temperatures below 600°C will be required to keep swelling rates under control. We noted also that cold-working the steel imparts about a factor of four swelling suppression under the range of conditions examined. Limitations imposed by tensile ductility also favor lower temperatures, with the greatest advantages shown for cold-worked material at the lowest temperature irradiated at 380°C and tested at 350°C.

Various attempts (10, 13–16) have been made to use available data on type 316 stainless steel to predict the lifetime of components of this material in fusion reactor service. These show that, depending on the assumptions made, any of a number of properties can be performance or lifetime limiting. The resulting lifetime-limits range from less than two years to unlimited reactor life. It is clear, however, from the composite picture that emerges from these treatments that attention will be required to the following points:

i. temperature below $\sim 550°C$ are required;

ii. lower temperatures will confer additional advantages;

iii. cold-worked material is preferable to solution annealed material;

iv. design precautions must be taken to keep stresses low; and

v. design allowable strains will be low.

It is also clear that a great deal more experimental data must be generated before definitive reactor designs and structural lifetimes can be achieved.

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