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THE RESPONSE OF AUSTENITIC STEELS TO RADIATION DAMAGE*

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Austenitic stainless steels are prominent contenders as first wall and blanket structural materials for early fusion power reactors. Properties affecting the performance of this class of alloys in the fusion irradiation environment, such as swelling, tensile elongation, irradiation creep, fatigue, and crack growth, have been identified. These properties and the effects of neutron irradiation on them are discussed in this paper. Emphasis is placed on the present status of understanding of irradiation effects.

1. INTRODUCTION

Breeder reactor materials programs have generated an extensive data base on irradiation effects in austenitic stainless steels, particularly on type AISI 316 stainless steel (316 SS). However, comparatively little is known about the effects of irradiation in environments which generate helium at rates expected in a fusion reactor. In this paper we describe the effects of irradiation on three important sets of properties (cavity swelling and creep, tensile strength and ductility, fatigue and crack growth) and summarize what is known about the effects of enhanced helium generation.

2. SWELLING BEHAVIOR

The cavity swelling phenomenon has dominated radiation damage studies for the past decade. Breeder reactor data at fluences of about ~100 dpa are now available on types 316 and 304 stainless steels and at lower fluences on the simpler Fe-Cr-Ni ternary alloys.[1-4] These data have recently allowed a comprehensive description of void swelling in austenitics as a function of both temperature and fluence. In complex alloys, swelling develops slowly at first, and follows a power-law dependence on dose. However, as a result of changes in dislocation and cavity sink strengths, microchemical changes and precipitate formation, swelling eventually accelerates into a rapid swelling regime characterized by a linear dependence on dose. This behavior is exemplified by the Fast Flux Test Facility (FFTF) first core heats of 316 SS (Fig. 1). For all temperatures in the range 500–650°C the low-swelling, or transient regime, extends out to ~40 dpa (1×10^{22} n/cm^2 ≅ 5 dpa). Swelling then accelerates until a

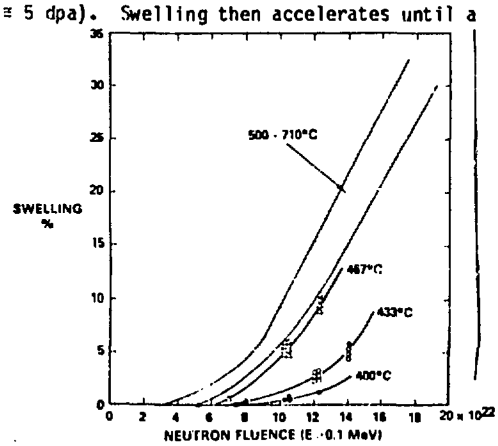

FIGURE 1
Swelling of 20%-cold-worked AISI 316 stainless steel irradiated in EBR-II. Four separate heats of FFTF first core steel are shown [H.R. Brager and F.A. Garner, J. Mater. 117 (1983) 159–76.

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rate of approximately 0.6%/dpa is achieved. Swelling values in excess of 30% have been observed without any reduction in swelling rate. The transient regime extends to progressively higher doses as the temperature is decreased below 500°C. The onset of the linear swelling regime is delayed by cold working and also by increasing the concentrations of certain minor alloying elements such as Ti, Si, and P. Similar behavior occurs in Fe-Cr-Ni ternary alloys. In alloys containing 15% Cr and 12–19% Ni, the transient regime is very short (~12 dpa) over the range 400–510°C, and the subsequent linear swelling rate approaches 1%/dpa (ref. 3). The extent of the low-swelling regime increases progressively with temperature above 510°C. Remarkable increases in the duration of the low-swelling regime occur when the chromium content is decreased or the nickel content is increased.[5]

This new perspective on the swelling behavior of austenitic stainless steels indicates that once the low-swelling or transient regime has ended, then unacceptably high-swelling rates will ensue over a wide temperature range. It is important therefore to understand the microstructural and microchemical changes which control the transition from low-swelling to the linear swelling regime, and to apply this knowledge to the development of alloys which will resist this transition.

The possible effects of increasing the He:dpa ratio are of major concern in the development of alloys for fusion reactor applications. Strong effects of helium injection or preinjection levels on cavity nucleation have been demonstrated in heavy-ion irradiations of 316 SS (refs. 6,7). While dual-ion beam irradiations provide a powerful means of studying radiation damage phenomena, a quantitative correlation with neutron irradiation behavior is confounded by surface-related phenomena and injected interstitial effects.[8] Further, the temperature and dose dependence of the segregation and precipitation behavior in complex alloys are not reproduced at the increased damage rate.[9] The only information available on helium effects at relevant damage rates is from comparisons of the behavior of 316 SS heats irradiated in EBR-II and in HFIR (where the He:dpa ratio is higher than that expected in a fusion reactor by a factor of 4 to 5). The most recent evidence indicates that increasing the He:dpa ratio from ~0.5 (EBR-II) to ~60 (HFIR) may extend or shorten the low-swelling regime, depending upon the heat chemistry (Fig. 2) (ref. 10). An extended low-swelling regime is associated with high number densities of bubbles, inhibition of radiation-induced segregation (RIS), and delayed conversion of bubbles to voids. A rapid transition into the linear swelling regime is associated with extensive RIS-affected precipitation and the rapid conversion of bubbles to matrix voids and precipitate-assisted voids. An example of these differences in cavity distribution is shown in Fig. 3 for a single heat of 20%-cold-worked (CW) type 316 SS irradiated in both the HFIR

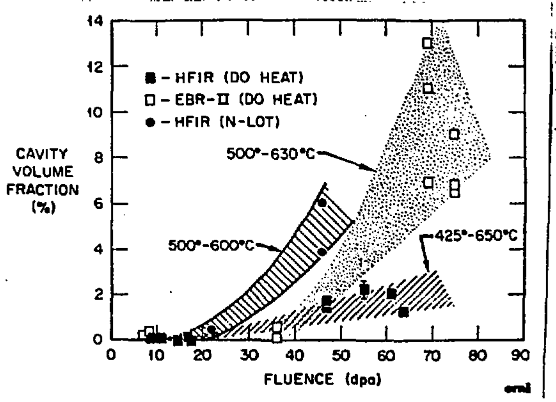

FIGURE 2
Swelling of 20%-cold-worked 316 SS. Data from two heats and irradiations from two reactors are shown to compare effects of He:dpa and heat chemistry (P. J. Maziasz, this volume).

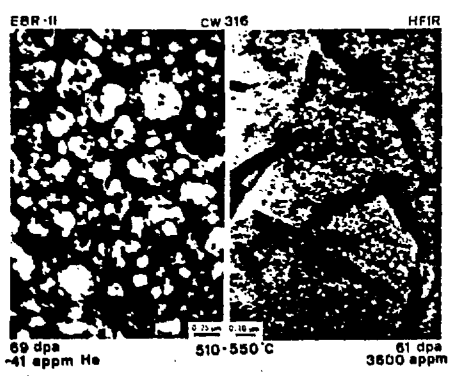

FIGURE 3
Microstructures of 20%-cold-worked 316 SS irradiated in EBR-II and HFIR. [P. J. Maziasz, J. Nucl. Mater. 108&109 (1982) 359–84.]

and EBR-II to approximately equal damage levels. A quantitative comparison of the cavity microstructures produced by irradiation in the two reactors and evidence for the effects of cavity number density on segregation and precipitation behavior are presented in ref. 10. It is not known whether or not the bubble-dominated microstructure can be maintained in HFIR beyond ~60 dpa. However, theory and modeling suggest that the transition to a linear swelling regime may occur more rapidly at the intermediate He:dpa ratios relevant to fusion reactors.[11]

The balance of evidence suggests that because of the limited low-swelling regime, the lifetime of components fabricated from 316 SS would be restricted to 40–50 dpa for temperatures >450°C. Swelling is unlikely to limit component lifetimes at temperatures below 300°C. Although swelling certainly occurs over the range 300–450°C, there are insufficient data to discern any effects of He:dpa ratio on the extent of the low-swelling regime.

The limited amount of published breeder reactor data indicates that the addition of titanium or niobium to the austenitic stainless steels prolongs the low-swelling regime[12,13] (Fig. 4). Titanium is thought to influence swelling behavior in several ways: (a) the residual gases are strongly gettered and are thus unavailable for stabilization of void nuclei; (b) the cold-work dislocation structure is stabilized and maintained to higher fluences; and (c) in environments with high He:dpa ratio the interaction between helium atoms and the MC particle-matrix interface strongly modifies the helium bubble distribution.[14]

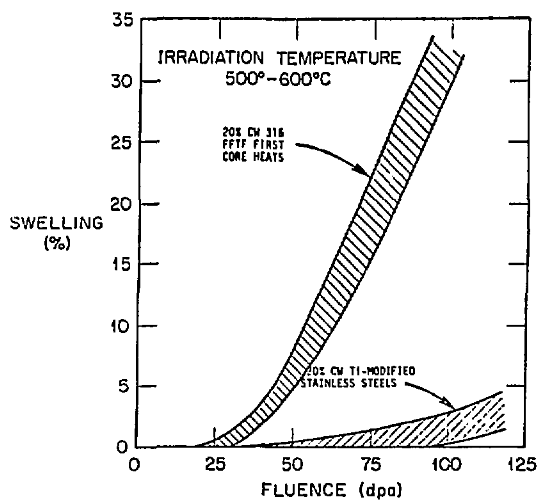

FIGURE 4
Swelling behavior of 20%-cold-worked 316 SS and titanium-modified variations irradiated in EBR-II at 500–600°C. [After B. A. Chin et al., Nucl. Technol. 57 (1982) 426–35.]

Figure 5 compares swelling behavior of 20%-CW 316 SS (N-lot) with developmental 25%-cold-worked PCA, a titanium-modified austenitic stainless steel with a composition as shown in Table I. Both materials were irradiated in the HFIR at 600°C to a maximum fluence of about 40 dpa and helium levels as high as 3000 at. ppm. The 316 SS has clearly emerged from the transient regime. In the PCA, however, the small

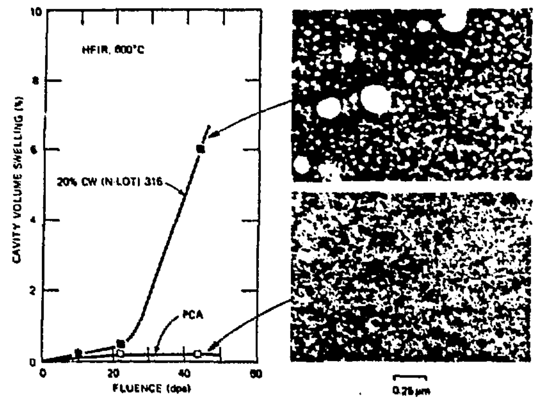

FIGURE 5
Swelling of 20%-cold-worked 316 SS steel (N-lot) and path A PCA (25%-cold-worked) (after P. J. Maziasz and D. N. Braski, this volume).

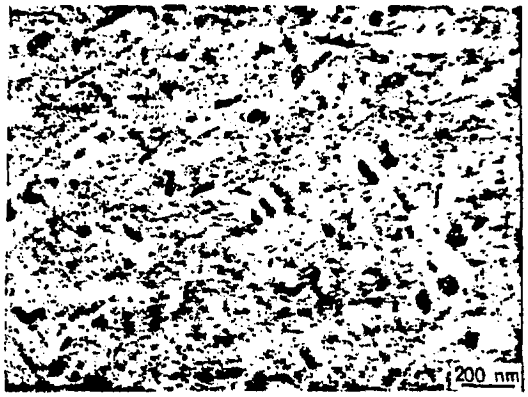

FIGURE 6
Fine bubbles in titanium-modified austenitic stainless steel irradiated to 40 dpa by dual-ion bombardment (20 at. ppm He/dpa) at 677°C [E. H. Lee, N. H. Packan, and L. K. Mansur, J. Nucl. Mater. 117 (1983) 123–33].

Table I. Alloy Composition (wt %)

316 MFE reference heat					
Cr	17.0	Mn	1.7	S	0.018
Ni	12.4	Si	0.67	P	0.037
Mo	2.1	Ti	<0.05	Fe	Bal
Path A PCA					
Cr	14.0	Mn	1.8	S	0.003
Ni	16.2	Si	0.4	P	0.01
Mo	2.3	Ti	0.24	Fe	Bal

bubbles attached to MC particles have resisted conversion to bias-driven voids.

Dual-ion beam studies on titanium-modified steels have demonstrated that when helium is generated at 20 at. ppm/dpa, a very fine dispersion of helium bubbles is nucleated in association with dislocation-nucleated MC particles (Fig. 6) (ref. 15). Because of the very high sink density, long-range diffusion of solutes coupled with defects is prevented, and the development of coarse precipitate phases through RIS is suppressed. Another important benefit of increasing the scale of bubble nucleation is that, for a given helium content, the average number of helium atoms per bubble is lowered. Thus, much higher doses are required before the number of helium atoms per bubble is sufficiently high for conversion to bias-driven voids to occur.[16] It is possible that other types of precipitates with different surface or interfacial properties may be even more effective in refining the scale or helium bubble nucleation and delaying the microstructural and microchemical changes which are the precursors of a linear swelling regime.[17] In the U.S. Alloy Development for Irradiation Performance program, a range of austenitic steels with compositions designed to promote fine stable particle dispersions is being studied. Parallel irradiations in the FFTF and the HFIR are being used to determine the effects of an enhanced helium generation rate on swelling resistance.

3. MECHANICAL PROPERTIES

In investigating the effects of the fusion radiation environment on the mechanical properties of materials, there are two basic considerations in determining the properties to be investigated and the types of tests. One is to identify a simple test that yields data from which mechanistic information can be extracted. The other is to identify specific properties to which the fusion reactor structure will be sensitive. The tensile test was selected on the basis of the first criterion. Properties in the second category are often discovered in design studies and are to some extent a function of the specific reactor design or the specific class of alloy. Certain properties have been selected because they appear to be critical to several designs and because there is general agreement on the necessity for experimental measurements. Examples of such mechanical properties are irradiation creep, fatigue, crack growth, and fracture toughness. Only the first three of these will be addressed in this paper. Because of the large specimen size required and the lesser importance of fracture toughness in austenitic steels, fracture toughness is often derived from other tests such as tensile tests.[18]

3.1 Tensile behavior

The effects of breeder reactor irradiation on the tensile properties of 20%-CW 316 SS are well established and the property changes correlated with changes in the microstructure.[19] For irradiation and test temperatures below ~500°C, the increase in yield strength with increasing dose saturates at ~15 dpa. For higher temperatures, the 20%-CW 316 SS softens initially but approaches a constant value of yield stress after ~10 dpa. See, for example, Fig. 7. Significant ductility is retained at all temperatures after irradiation to doses of 40–50 dpa. Early tests of HFIR-irradiated material revealed tensile ductilities of only a few tenths of a percent at 575°C for material irradiated to 75 dpa and containing 4300 at. ppm He.[20] More recent tests on the MFE reference

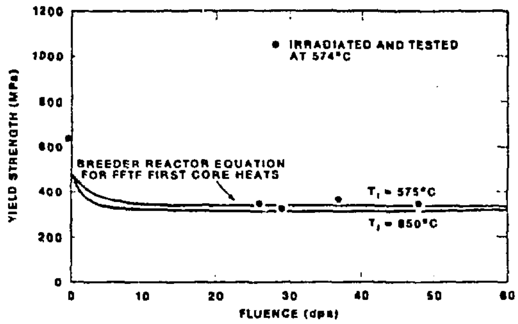

FIGURE 7
Yield strength of 20%-cold-worked 316 SS (data points are for MFE reference heat irradiated in HFIR). The curves are for FFTF first core steel irradiated at 575 and 650°C.

heat of 316 SS have exhibited total elongations in excess of 8% at 575°C for material irradiated to about 50 dpa and containing approximately 3000 at. ppm He, as shown in Fig. 8 (ref. 21). Also shown in Figs. 7,8 are curves from tests of the FFTF first core heat of 316 SS irradiated in EBR-II showing good agreement with the HFIR data. There are sufficient data from the fast reactor program to separate the dependence on test temperature and irradiation temperature. At a test temperature of 575°C, there is a very weak dependence of tensile properties on the irradiation temperature when it is increased beyond the test temperature. Even though there is evidence to indicate that the HFIR irradiation temperatures were higher than stated,[22] the weak dependence on irradiation temperature allows a valid comparison of tensile properties to be made. Comparison of data for test temperatures of 350, 450, and 575°C shows that both ductility and strength behavior of HFIR-irradiated material are similar to those found on other heats of 316 SS irradiated in fast reactors where only trace amounts of helium are present. However,

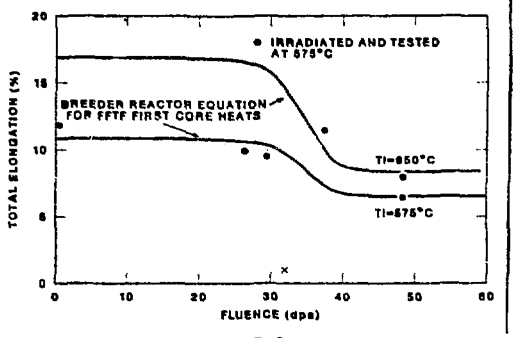

FIGURE 8
Total elongation of 20%-cold-worked 316 SS (data points are for MFE reference heat irradiated in HFIR). The curves are for FFTF first core steel irradiated at 575 and 650°C. The data point designated by **x** is for material tested at 675°C.

very limited data show ductilities ≈1% for material irradiated in the HFIR (30 dpa, 2000 at. ppm He) at 575°C and tested at 675°C (Fig. 8). Except for temperatures in excess of 600°C and especially low strain rates, the tensile behavior of HFIR-irradiated material is similar to that of fast-reactor-irradiated material.

3.2 Irradiation creep

The phenomenon of irradiation creep in breeder reactor-irradiated 20%-CW 316 SS has been well characterized.[23] A transient period is followed by a slowly increasing creep rate with increasing fluence (Fig. 9). There is an essentially linear dependence of irradiation creep rate on stress and a very weak dependence upon temperature. The rapid creep rates at the higher temperatures in Fig. 9 are due mostly to thermal creep. The dependence on flux has been studied by Lewthwaite and Mosedale in the Dounreay Fast Reactor.[24] Their data (after correction for temperature dependence using an activation enthalpy of 0.114 eV derived from proton irradiation data)[25] are shown in Fig. 10. From this figure, creep rate appears to be an inverse function of damage rate. There is very little information on the effects of helium generation rate on irradiation creep. The most

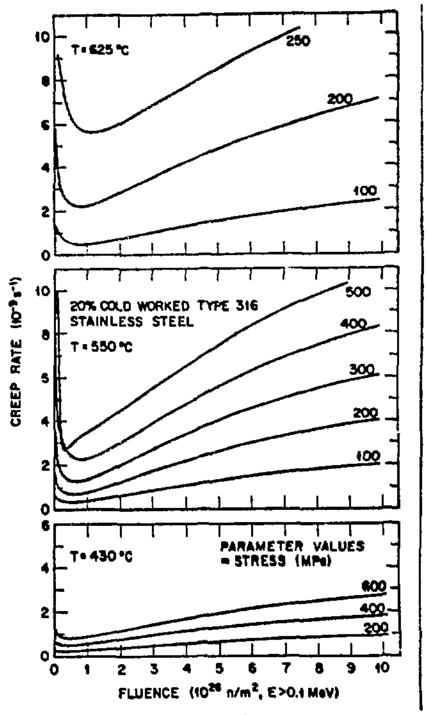

FIGURE 9
Creep rate for 20%-cold-worked AISI 316 stainless steel (FFTF first core heat) given by Puigh et al., ASTM STP 782, 1982, pp. 108–21.

relevant data at the present time are from pressurized tubes of both 316 SS and the MFE program Path A PCA irradiated in the ORR in an experiment where the neutron spectrum is tailored to achieve the fusion reactor He:dpa ratio typical of fusion reactor irradiation.

Figure 11 shows data from an interim examination at 5 dpa where a helium level of about 50 at. ppm was achieved. At 330°C no difference is apparent between the ORR data and the fast reactor data from another heat of 316 SS (FFTF first core). At 500°C the ORR data fall significantly below the fast reactor data. However, at this early stage of the experiment this difference in behavior could equally well be related to differences in heat chemistry. Both higher exposures and control experiments using

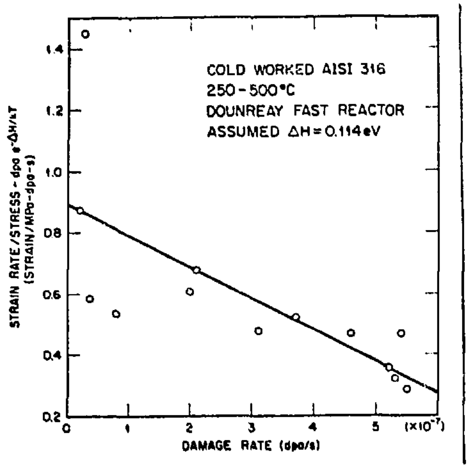

FIGURE 10
Irradiation creep strain rate per unit stress per dpa in cold-worked AISI 316 stainless steel showing flux dependence. The data were adjusted for temperature using an activation enthalpy of 0.114 eV [data from G. W. Lewthwaite and D. Mosedale, J. Nucl. Mater. 90 (1980) 205–15].

the same steel irradiated in fast reactors are in progress.

3.3 Fatigue properties

Fatigue is a property that was clearly recognized early in fusion reactor research as an area of concern in cyclic operating tokamak reactors. Even though the design trend is toward progressively longer burn times or even steady state, fatigue is still a concern because of startup and shutdown as well as plasma disruption events. Again, the data most relevant to a helium-producing irradiation environment are from irradiations in the HFIR. Specimens of 20%-CW 316 irradiated to damage levels as high as 15 dpa and levels of helium up to 900 at. ppm irradiated at 430 (ref. 26) and 550°C (ref. 27) have been examined in fatigue. The 430°C irradiation and test temperature data exhibit a reduction in fatigue life by a factor of 3 to 10 (Fig. 12); however, at 550°C, no reduction in fatigue life is observed (Fig. 13). Even raising the test temperature 100°C above the irradiation temperature to 650°C fails to reduce fatigue life over the unirradiated material.[28] Although these tests at low doses have not revealed any significant effect of irradiation on fatigue life, it has been shown that under other testing conditions, helium may have serious effects on mechanical properties. For example, Batra et al.[29] demonstrated a frequency

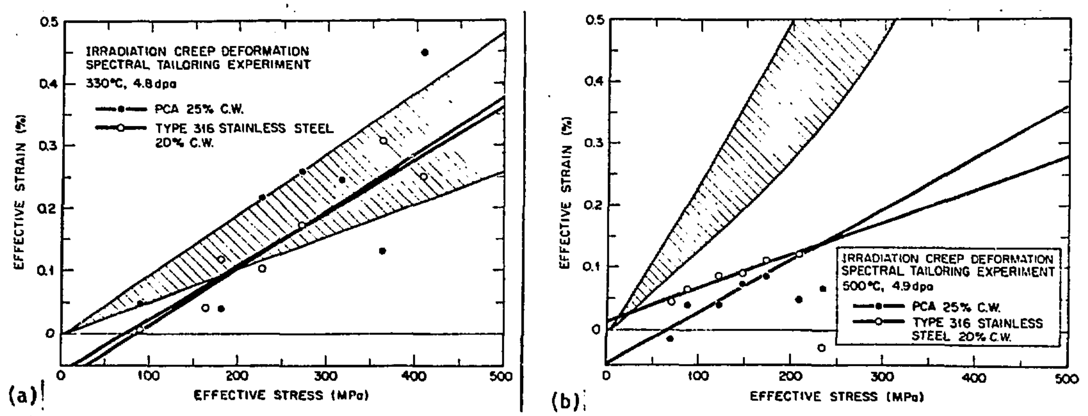

FIGURE 11
Irradiation creep in austenitic stainless steels irradiated in the ORR in a spectrally tailored experiment designed to achieve the fusion He:dpa value. The cross-hatched regions show fast reactor data for FFTF first core 316 reported by Puigh et al., ASTM-STP-782. (a) 330°C and (b) 500°C.

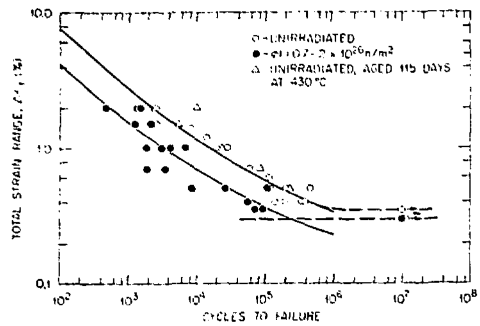

FIGURE 12
Fatigue life of 20%-cold-worked 316 SS irradiated in the HFIR (6–15 dpa, 200–900 at. ppm He) at 430°C and tested at 430°C.

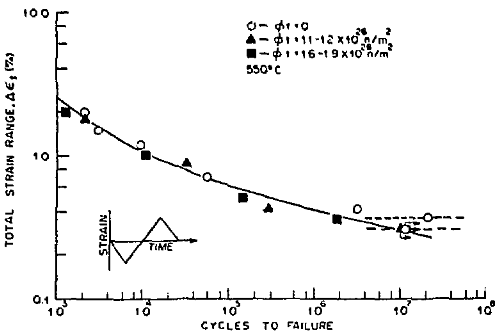

FIGURE 13
Fatigue life of 20%-cold-worked 316 SS irradiated in the HFIR (9–15 dpa, 400–800 at. ppm He) at 550°C and tested at 550°C.

dependence on the fatigue life of 316 SS preinjected with 800 at. ppm He at 600°C; a reduction in fatigue life by a factor of ~50 occurred when the test frequency was reduced from 5 to 0.5 Hz. The work of Van der Schaaf et al.[30] demonstrated that irradiation of AISI 304 to very low fluences resulted in reductions in ductility in low strain rate creep tests at 600°C. They ascribed this ductility loss to helium-enhanced intergranular crack growth. Clearly, these results indicate that the effects of helium become more significant as the strain rate is lowered. The sensitivity of the fatigue test to helium effects would be increased by interjecting a tensile hold period into the fatigue cycle. This mode of testing would also be more relevant to the operation of a tokamak fusion reactor than the conventional sawtooth or sinusoidal fatigue cycle.

Data on fatigue crack growth rates are limited to very low levels of displacement damage. The highest dose data available is that of Michel and Smith[31] for 316 SS irradiated in EBR-II to ~11 dpa. No significant effects of irradiation on crack growth rate were observed for both annealed and 20% CW 316 SS irradiated and tested at 427°C. However, the 20%-CW 316 SS exhibited an irradiation-induced increase in crack growth rate by about a factor of 10 when similarly irradiated specimens were tested at 593°C. When a hold period was introduced into the fatigue cycle, an acceleration of crack growth rate was observed for both the annealed and 20%-CW conditions. There are no data on the effects of helium on crack growth. However, it is expected that any reduction in fracture toughness produced by irradiation would result in an acceleration of fatigue crack growth.[18]

4. SUMMARY

1. The transient swelling regime for 20%-CW 316 SS is affected by helium generation rate and may be extended or shortened depending upon heat chemistry. Present evidence suggests that cavity swelling will probably limit the application of this material to fluences of 40–50 dpa for operating temperatures >300°C.

2. The titanium-modified steels exhibit greatly improved swelling resistance under breeder reactor conditions. Furthermore, there is strong evidence that helium concentrations of 2000–3000 ppm can be accommodated in a dispersion of fine bubbles attached to MC particles.

It is thought that further development of these steels could result in microstructures which maintain a low swelling rate mode to fluence levels approaching 100 dpa in environments with high He:dpa ratios.

3. Although the irradiation creep behavior of both 316 SS and titanium-modified steels is well characterized, the effects of helium generation rate on creep have yet to be determined.

4. The tensile ductility of 20%-CW 316 SS is not seriously degraded by neutron irradiation up to ~50 dpa at temperatures ≤575°C even when high levels of helium (~3000 ppm) are generated. At these helium levels, serious embrittlement is expected to occur at temperatures >600°C.

5. Fatigue life of 20%-CW 316 SS is not seriously impaired by irradiation to ~15 dpa at 430–550°C even in the presence of >900 at. ppm He. It is recommended that a more appropriate testing mode to assess helium effects in austenitic stainless steels would be to interject tensile hold periods into the fatigue cycle.

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