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MECHANICAL PROPERTIES OF MATERIALS IN FUSION REACTOR FIRST-WALL AND BLANKET SYSTEMS*

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With respect to the effects of irradiation on mechanical properties, the most significant difference between fast fission and fusion reactor spectra is the relatively large amount of helium produced by (n, α) transmutations in the latter. Relevant information on the effects of large amounts of helium (with concomitant displacement damage) comes from irradiation of alloys containing nickel in mixed spectrum reactors. At helium levels of interest for fusion reactor development, properties are degraded to unacceptable levels above $T_m/2$. Below this temperature, strength and ductility are retained and fractures remain transgranular. Importantly, the properties remain sensitive to composition and structure. A comparison of the response of bcc refractory alloys to that of stainless steel at equivalent damage levels shows the same general trends in properties with homologous temperature. The refractory alloys do offer potential for higher temperature applications because of their higher melting temperatures.

1. INTRODUCTION

In presently envisioned fusion reactors, the relatively high heat fluxes through the first wall coupled with pulsed operation causes physical properties such as thermal conductivity, thermal expansion and modulus to be of great importance in the evaluation of materials performance. Equally important are the mechanical properties associated with cyclic stresses such as fatigue and crack growth. Materials selection for design studies has been based almost solely on these considerations. Other considerations such as fabricability, availability, and compatibility with breeding and cooling fluids, must also be factored into the materials selection equation. When this is done the list of available materials with potential application includes the austenitic stainless steels, nickel base alloys, titanium alloys, refractory metal alloys such as those based on vanadium or niobium, and possibly the ferritic steels.

The most difficult aspect of any evaluation of the potential of these alloys for use in the high neutron flux region is the very limited knowledge of irradiation damage in the fusion reactor neutron spectrum. The major difference in the fusion spectrum as compared to a fission reactor is the high energy neutrons (up to 14.1 MeV). No major differences have been found in the basic form of displacement damage (i.e., defect cluster size and type) produced by these neutrons, and it appears that the amount of displacement damage (dpa) can be calculated to within a factor of 2 [1-3]. The much higher production of transmutation products (particularly helium and hydrogen) in the fusion spectrum (Table I) is the primary difference. It is

Table I. Radiation damage in alloys typical of the alloy systems proposed for fusion first-wall applications

Metal or Alloy	Per MWy/m ²			
	dpa	at. ppm H	at. ppm He	
Aluminum	14	296	316	
Ti-6 Al-4 V	16	175	142	
Ferritic steel (Sandvik HT-9)	11	450	110	
Austenitic SS (316)	11	532	147	
Nickel-base alloy (Nimonic PE-16)	12	780	240	
V-15 Cr-5 Ti	11	245	47	
Niobium	7	105	29	

important to note that large amounts of helium are produced in all alloys. All relevant information on the effects of large quantities of helium in the presence of displacement damage comes from irradiation of alloys containing nickel in mixed spectrum reactors. For other alloys the data are limited to hardening effects in the "absence" of helium and to effects, in most instances, of relatively small amounts of helium introduced by accelerator injection or the tritium trick. This paper will first review the most important findings in alloys containing nickel with particular emphasis on high helium and dpa levels. This will be followed by an examination of results for other alloy systems (basically displacement and helium effects separately) and an attempt to analyze these results for similarities and differences with alloys containing nickel.

2. ALLOYS CONTAINING NICKEL

At high helium contents, data are limited to uniaxial tensile and creep-rupture response.

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Expected trends in fatigue and fatigue crack growth properties in the fusion reactor environment must be inferred from tensile and creep-rupture properties at high helium contents, properties of the unirradiated alloys, and the observed effects of fast reactor irradiation (in which case the He:dpa ratio is much below that of a fusion reactor).

Figure 1 shows the yield strength and elongation for annealed 316 stainless steel. The yield stress data reported by Bagley et al. [4] and DuPouy et al. [5] is for damage levels of approximately 50 dpa, while that reported by Fahr et al. [6] is for 6-12 dpa. The increase in yield strength which results from displacement damage tends to saturate at about 10 dpa. Agreement between these data sets is thus expected. Data from Bloom and Wiffen [7] are for a displacement level of about 50 dpa, but helium contents of about 4000 at. ppm (equivalent to 25 MWy/m²). At low temperatures, near 300°C, the increase in yield strength is nearly the same for irradiations in which the He:dpa production is high or low. As the temperature increases, the yield strength for high He:dpa falls below that for low He:dpa. Above 650°C the samples containing large amounts of helium fail in the elastic strain region of the stress-strain curve at stresses below the expected yield stress.

Total elongation of annealed 316 stainless steel as a function of temperature for tests in which the irradiation and test temperatures are approximately equal is also shown in Fig. 1. At all temperatures in the range 300-700°C the reduction in ductility is significantly larger at helium levels anticipated for fusion reactor systems than at levels produced in a fast reactor. Data are very limited for fast reactor irradiations (low He:dpa ratio) at high damage levels (near 50 dpa). At low temperatures

(300°C) Bagley et al. [4] report approximately 16% total elongation at 50 dpa. In comparison, Bloom and Wiffen [7] report less than 1% total elongation at the same dpa level but at a helium content of ~4000 at. ppm. No data have been published for irradiation at higher temperatures to high damage levels and low He:dpa ratios. At 6-10 dpa Fahr et al. [6] report total elongations of 15-26% for temperatures ≤700°C. For irradiations at a low temperature (300°C) and tests in the range 600-700°C Bagley et al. [4] show elongations in the range 20-40% at about 50 dpa. It thus appears that annealed 316 stainless steel retains reasonably high ductility for fast reactor irradiations to high dpa or fluence levels and helium contents near 20 at. ppm. At the same dpa level but high He:dpa ratios (up to 4000 at. ppm He) the ductility is in the 0.6-2.6% range for temperatures to 575°C. At higher temperatures the total plastic elongation drops to zero [7].

Figure 2 shows yield stress and total elongation of 20%-cold-worked 316 stainless steel. The curve labeled low He:dpa was drawn through the data of Fahr et al. [6] for samples irradiated to 5-11 dpa (1.5-3 at. ppm He/dpa). Data at 40-50 dpa can only be estimated. Barnaby et al. [8] report percent reduction in yield strength as a function of temperature for damage levels of 7, 16, 30 and 44 dpa. At 16 dpa their results are in agreement with those of Fish and Watrous [9]. These percent reductions were applied to the unirradiated yield strength values reported by Fahr et al. [6]. The material used by Fahr et al. for the fast reactor irradiations was also used by Bloom and Wiffen for the mixed spectrum reactor irradiations to high helium contents. The estimated yield strengths shown in Fig. 2 result. For irradiation to about 50 dpa and 4000 at. ppm He, the yield strength is lower than estimated values for the low He:dpa

Fig. 1. Yield Strength and Total Elongation of Annealed 316 Stainless Steel in the Unirradiated Condition and after Irradiation in a Fast Reactor (Low He:dpa) and a Mixed Spectrum Reactor (High He:dpa Ratio).

Fig. 2. Tensile Properties of 20% Cold-Worked 316 Stainless Steel Irradiated in Neutron Spectra Which Produce Low He:dpa (~ 0.3 at. ppm He/dpa) and High He:dpa (~ 100 at. ppm He/dpa) Ratios at 40–50 dpa.

fast reactor irradiations. Total elongation also appears to be reduced to lower levels for high He:dpa than for low He:dpa ratios. Bloom and Wiffen [7] report total elongations of 3.8% at 350°C and 50 dpa and 4000 at. ppm He compared to 10% reported by Barnaby et al. [8] at 50 dpa and ~ 20 at. ppm He. At this helium content, ductility was reduced to zero at 575°C and higher.

In Figs. 2 and 3 the data for low He:dpa ratio reported by Barnaby et al. [8] and Bagley et al. [4] is for tests conducted at $1.6 \times 10^{-3} \text{ s}^{-1}$. Fahr et al. [6] employed a strain rate of $3.3 \times 10^{-5} \text{ s}^{-1}$ and the data at high He:dpa ratio [7] is for a strain rate of $5 \times 10^{-5} \text{ s}^{-1}$. These differences make it difficult to compare the detailed response. Fish [9], however, has found that below about 550°C the ductility of cold-worked 316 stainless steel irradiated to about $3 \times 10^{26} \text{ n/m}^2$ ($> 0.1 \text{ MeV}$) is essentially independent of strain rate in the range $2 \times 10^{-5} \text{ s}^{-1}$ to $3.5 \times 10^{-2} \text{ s}^{-1}$. Thus, below 550°C the conclusions are certainly valid. Above 550°C the ductility decreases with decreasing strain rate. However, at least for fluences to $\sim 3 \times 10^{26} \text{ n/cm}^2$ ($> 0.1 \text{ MeV}$), there is measurable ductility (i.e., failure does not occur in the elastic range). Thus, over the temperature range of interest, irradiations which produce high He:dpa ratios cause significantly larger reductions in ductility than irradiations which produce low He:dpa ratios when compared at about the same dpa.

Fig. 3. Effect of Irradiation to Approximately 10 dpa and 1700 at. ppm He on the Ductility and Fracture Behavior of 20%-Cold-Worked Inconel 600, ref. [10].

Trends in tensile properties with damage level are shown in Table 2. At 350°C the yield and ultimate tensile strengths initially increase and then decrease slowly with damage level. The true fracture stress decreases and the uniform elongation increases with fluence. Total elongation remains at an acceptable level to very high damage levels. At 575°C, however, all strength and ductility properties decrease monotonically with damage level. A further important observation is that at 350°C the tensile fractures remain transgranular, whereas at high temperatures they are clearly intergranular. At temperatures at which He causes reduced strength and ductility and pronounced intergranular fracture, a marked reduction in creep-rupture properties also occurs. At 550°C and 310 MN/m² the rupture life is reduced by over five orders of magnitude [7].

The only other alloy for which mechanical property data at high helium concentrations has been published is Inconel 600 [10]. Property changes as a function of irradiation and test temperature are similar to those reported for 316 stainless steel except that the properties of Inconel 600 were similar when irradiated in the annealed or cold-worked conditions. The changes in fracture mode are also similar to those observed in 316 stainless steel and are illustrated in Fig. 3. At low temperatures (300°C) fractures are of a ductile transgranular nature — the reduced total elongations resulting largely from reduced uniform strain. At 500°C grain boundary fracture accounts for about 50% of the fracture surface and at 700°C the fractures are entirely intergranular.

2.1 Fatigue and fatigue crack growth (FCG)

Design studies have generally concluded that fatigue and fatigue crack growth are the mechanical properties which will control structural lifetime in reactors which have a cyclic burn [11]. The importance of these properties stems from the cyclic heat load, and thus cyclic thermal stress, in the first-wall structure. Fatigue failures involve both the initiation and propagation of a fatigue crack. Fatigue crack growth is, in reality, the last stage in the fatigue life. The growth rate can be correlated with the stress intensity factor as calculated using linear-elastic fracture mechanics (LEFM) techniques. Lifetime predictions based on FCG assume the preexistence of flaws of a given size. In stainless steels and nickel-base alloys, the mode of cracking and the crack propagation rates are sensitive to many variables. At low temperatures, in the regime where deformation is athermal, initiation and propagation are generally transgranular [12]. As the temperature is increased and/or the strain rate is reduced, a transition to intergranular initiation and propagation occurs. The temperature at which this transition in fracture mode occurs is near $T_m/2$ or about 550°C for austenitic stainless steels. A similar transition in fracture mode occurs in

tensile and creep fractures. Helium tends to reduce the temperature at which the transition from transgranular to intergranular fracture occurs.

James [12] has recently published a detailed review of fatigue-crack propagation behavior in austenitic stainless steels, and Michel and Korth [13] have reviewed available information on the effects of irradiation. One of the most important general observations is that fatigue- and fatigue crack growth rates are sensitive to the environment. At elevated temperatures, crack growth rates decrease when the test environment is changed from water or steam to air to inert environments such as helium, argon, and liquid sodium (Fig. 4). In the aggressive environments, crack growth appears to increase with temperature while in the inert environments growth rates at elevated temperature are approximately the same as at room temperature. Since these effects can be large (i.e., a factor of 10 higher crack growth rate at 538°C in air than in liquid sodium), it is important to analyze material behavior in the environment appropriate for the application.

For test conditions which produce transgranular fracture, the crack growth rate is not strongly dependent on microstructural features, nor is it altered by neutron irradiation at least to

Fig. 4. Effect of Test Environment on Fatigue Crack Growth in Annealed Type 304 Stainless Steel at 538°C. After James [12].

Table 2. Tensile Properties as a Function of Fluence and Helium Concentration for 20%-Cold-Worked Type 316 Stainless Steel [7]

Irradiation Temperature (°C)	Fluence >0.1 MeV (n/m ²)	Damage (dpa)	Helium (appm)	Stress, psi ^a		True Fracture	Elongation, %	
				Yield	Ultimate		Uniform	Total
Test Temperature 350°C								
	0	0	0	131 100	136 700	195 600	0.6	5.1
385	4.1 × 10 ²⁶	56	1660	146 500	150 600	155 800	0.8	2.1
380	7.1 × 10 ²⁶	97	4020	109 100	115 800	135 500	3.3	3.8
Test Temperature 575°C								
	0 ^b	0	0	112 000	117 000	140 000	0.9	6.0
	0 ^c	0	0	81 000	89 000	128 000	1.7	9.0
590	2.2 × 10 ²⁵	3.0	16	76 200	85 200	87 000	2.5	2.5
600	4.0 × 10 ²⁵	5.5	33	73 900	78 300	78 500	0.5	0.5
575	5.0 × 10 ²⁵	6.8	48	60 300	63 100	63 500	0.5	0.5
605	4.1 × 10 ²⁶	56	1660	30 300	41 000	41 500	1.5	1.5
600	8.7 × 10 ²⁶	119	5940	46 900	46 900	47 000	0.2	0.2

^aTo convert to megapascals, multiply by 0.006895.

^bInterpolated from test results at 550 and 650°C on material tested in the 20%-cold-worked condition.

^cInterpolated from test results on 20%-cold-worked samples that had been aged 4000 h at 550 and 650°C and tested at the aging temperature.

relatively low dpa and helium levels. Aging annealed 304 or 316 stainless steel at either 538 or 549°C for times to 6000 h reduced the crack growth rate at low ΔK levels ($\sim 10 \text{ MN/m}^{3/2}$) by approximately a factor of 2. With increasing ΔK the effects of aging disappeared. Cold working 304 or 316 stainless steel tends to reduce the crack growth rate relative to the annealed condition. Aging the cold-worked structure tends to increase the crack growth rate relative to the cold-worked condition. The effects of neutron irradiation on crack growth have only been examined for dpa and helium levels which are very low when compared to expected fusion reactor requirements. For the damage levels investigated and in the transgranular fracture regime the crack growth rates are either unchanged or increased at low ΔK and reduced at high ΔK . The magnitude of the observed effects can be seen from the results of Shahinian et al. [14] shown in Fig. 5. For annealed 316 stainless steel irradiated at 288°C in a thermal reactor ($1.8 \times 10^{25} \text{ n/m}^2$, >0.1 MeV) fatigue crack growth rates were reduced at all ΔK values. Irradiation in a fast reactor at about 400°C to $1.2 \times 10^{25} \text{ n/m}^2$ (>0.1 MeV) slightly increased the growth rate at low ΔK and decreased it at high ΔK . Similar trends were observed for annealed 304 stainless steel. The improvement in crack growth properties is attributed to the increased yield strength resulting from the irradiation and thus a smaller plastic zone at the crack tip.

Crack growth generally becomes more sensitive to irradiation and to metallurgical variables at temperatures at which the fracture mode tends to be intergranular. Samples irradiated at the same conditions described above were also tested at 593°C. The crack growth rates generally increased with irradiation as is shown for 304 stainless

steel in Fig. 6. The displacement damage was recovered prior to testing. The test temperature is at the lower edge of the temperature range where helium begins to influence properties strongly. When a hold time is introduced into the tensile portion of the cycle at elevated temperatures (e.g., 593°C, Fig. 7), crack growth

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Fig. 5. Effect of Irradiation on fatigue crack growth in annealed 316 stainless steel when tested below $T_m/2$. Tensile fractures are transgranular for these conditions. After Shahinian et al. [14].

Fig. 6. Effect of Irradiation on Fatigue Crack Growth in Annealed 304 Stainless Steel Tested at 593°C. After Shahinian et al. [14].

Fig. 7. Effect of Hold Time in the Tensile Portion of the Cycle in Fatigue Crack Growth in 20% Cold-Worked 316 in the Unirradiated Condition and After Irradiation in EBR-II. After Michel and Korth [13].

rates are generally increased [13]. Irradiation produces a further increase in crack growth rate, but the increase does not appear to depend on hold time. In contrast, a hold time and a hold time plus irradiation had little effect on crack growth rate at 427°C which is in the transgranular fracture regime.

Similar trends regarding the effects of irradiation are observed in high cycle fatigue as in fatigue crack growth — increased life at low temperatures and decreased life at high temperatures. Michel and Smith [15] examined the high-cycle strain-controlled fatigue life of 304 stainless steel irradiated to maximum fast neutron fluences of $1.5 \times 10^{27} \text{ n/m}^2 (> 0.1 \text{ MeV})$. At both 427 and 593°C test temperatures, the fatigue life decreased with increasing irradiation temperature. However, for irradiation temperatures below about 450°C the fatigue life is higher for irradiated than unirradiated material (Fig. 8). Brinkman et al. [16] and Broomfield et al. [17] found that for irradiation and test temperatures $> 700^\circ\text{C}$ the fatigue life of austenitic stainless steels is reduced by irradiation. The magnitude of the reduction decreased with decreasing strain amplitude. Near the endurance limit, irradiation appeared to have little effect. There have been no systematic investigations of the effects of irradiation on the fracture mode. Work in progress by Michel [18] indicates that at a test temperature of 427°C fractures remain transgranular, while at 593°C the fractures are generally intergranular.

3. REFRACTORY AND REACTIVE METAL ALLOYS

Refractory metals with bcc structure have basic physical properties which make them attractive for fusion reactor applications. As shown in Table 3, the relatively high thermal conductivity coupled with a low coefficient of thermal expansion and modulus results in lower thermal stresses (for equivalent heat fluxes) than in austenitic stainless steels. The higher melting points of the refractory metal alloys would allow them to be used at a lower fraction of their absolute melting point for a given reactor design (i.e., coolant temperatures, wall loading, etc.). These alloys could possibly be used at sufficiently low fractions of T_m that thermally activated deformation and fracture processes could be avoided. Titanium alloys also have a combination of physical properties which give rise to low thermal stresses. These alloys are basically low temperature alloys, however, having maximum use temperatures in the neighborhood of 500°C.

The amount of information available on the most important mechanical properties is very limited. In contrast to those alloys containing nickel, there are no data available at helium contents relevant to the fusion reactor (see Table 1) in the presence of displacement damage. From available results we can only identify those properties which we expect to be degraded by the fusion reactor neutron spectrum and may limit alloy performance. It is impossible to determine

Fig. 8. Effect of irradiation on the fatigue life of annealed type 304 stainless steel at 427 and 593°C test temperatures. From Michel and Smith [15].

Table 3. Physical Properties* of Various Alloys Proposed for Fusion Reactor Application

Alloy	Thermal Conductivity W/m °K (K)	Thermal Expansion 10 ⁻⁶ /°K (α)	Young's Modulus GN/m ² (E)	Thermal Stress Coefficient (αE/k)
316	22	18.3	160	133
Ti-6Al-4V	14.0	10.0	72	51
V-20Ti	26.0	10.0	112	43
Nb-1Zr	31.0	7.9	65	17

*Properties at 773 K.

temperature and fluence ranges within which acceptable properties will be maintained.

In general, irradiation produces changes in strength properties of the bcc alloys which have a temperature dependence similar to that found in fcc alloys. At low temperatures, below 0.5 T_m , the irradiation-produced damage structure, dislocation loops and voids, cause a large increase in yield stress. As the temperature is increased the irradiation damage structure coarsens and the increase in yield stress is reduced. Above 0.5–0.5 T_m , the yield stress in the irradiated condition is nearly equal to that in the unirradiated condition. A major difference between fcc and bcc alloys is the way in which the changes in strength influence important ductility and fracture mechanisms.

For low homologous irradiation and test temperatures the yield stress may be raised above the cleavage stress and brittle failure results. In molybdenum-based alloys, including Mo, Mo-0.5 Ti and Mo-50 Re, irradiation at temperatures in the range 400–600°C to fluences of $1.5\text{--}6.1 \times 10^{22}$ n/cm² (producing 6.5–23 dpa and in all cases less than 1 at. ppm He) raised the DBTT to near 550°C at a strain rate of 0.02 min⁻¹ [19]. As the strain rate was reduced the DBTT (or the

increase in DBTT) was reduced. At 400°C, Mo-0.5 Ti showed brittle failures at strain rates of 0.02 and 0.002 min⁻¹, but unstable ductile failures at 0.002 min⁻¹. The helium contents in these experiments were far below those which will be produced in a fusion reactor. In alloy systems where results are available it appears that helium causes irradiation defects (both loops and voids) to be produced on a much finer scale than in irradiations in which the He:dpa ratio is low as in fast reactors [10,20]. These defects are responsible for the flow stress being raised above the cleavage stress. In the fusion-reactor spectrum the shift in DBTT to temperatures above room temperature and into the operating range may well occur at irradiation temperatures higher than indicated by fast reactor results.

A second property change which is a direct result of the irradiation-produced defect structure is the reduction in uniform strain. This occurs in both fcc and bcc alloys, but in some bcc alloys the uniform strain is reduced to zero [19]. When this occurs the alloys have no ability to work-harden. In an engineering structure, deformation would become concentrated and quickly exceed the failure strains. The clearest example of zero uniform strain is the alloy Nb-1 Zr. Tensile curves for irradiated samples are shown in Fig. 9. Although the total elongation values are acceptable at this damage level, the uniform strain is zero—necking begins immediately on yielding. Similar behavior occurs in molybdenum alloys, when tested at strain rates and temperatures at which brittle fracture does not occur, and in the tantalum alloy, T-111.

The transition from ductile to brittle fracture and the reduction in uniform strain discussed in the previous paragraphs result from the effects of the irradiation-produced defect structure on flow properties. It is often stated that the bcc alloys are more resistant to embrittlement

Fig. 9. Effect of irradiation on the stress-strain behavior of Nb-1 Zr. After Wiffen [19].

by helium (which occurs above $v T_m/2$) than are the austenitic stainless steels and nickel-base alloys. It is questionable whether this generalization can be supported by the available data. In neutron irradiation experiments conducted to date the transmutation-produced helium contents in the refractory metals are not more than a few at. ppm (usually less than 1 at. ppm). In most instances the ductilities remain high. If we compare the response of refractory alloys and stainless steels at equivalent damage levels and homologous temperatures the high-temperature behavior appears remarkably similar. Figure 10 shows a total elongation ratio (irradiated:unirradiated) as a function of homologous temperature. For annealed 304 stainless steel containing 30 at. ppm He [21] (cyclotron-injected) the ratio begins to decrease above $0.5 T_m$. Results of Mattas et al. [22] for annealed V-15 Cr-5 Ti also containing 30 at. ppm He (introduced by the tritium trick) show a reduction in high-temperature ductility beginning at a somewhat lower homologous temperature. The combined effects of displacement damage and transmutation-produced helium are shown for annealed 304 stainless steel [23] and annealed VANSTAR-7 [19] at approximately equal damage levels. The fracture mode of the V-15 Cr-5 Ti alloy at elevated temperatures changed from transgranular to intergranular with the introduction of 30 at. ppm He. Ehrlich and Böhm [24] found that 1-1.5 at. ppm He produced significant reductions in tensile ductility of three V-Ti-Nb alloys at 850 and 950°C but no effect at 650 and 750°C. A fourth

alloy, V-3 Ti-1 Si, showed no loss of ductility at temperatures to 900°C at this helium level. When reduced ductility was observed, it was clearly associated with intergranular fracture.

Atteridge et al. [25] have introduced helium in concentrations up to about 500 at. ppm into niobium by the "tritium trick." Tensile properties were determined at 1020°C ($0.48 T_m$). Strength properties increased and ductility decreased with increasing helium concentration. With no helium, the fractures were transgranular-ductile with extensive matrix deformation near the fracture. At 500 at. ppm He, ductility was markedly reduced - no highly deformed grains in the necked region, and there was significant grain boundary decohesion.

The above results suggest that the bcc refractory metal alloys (at least those alloys of vanadium and niobium examined to date) do not respond in an inherently different manner to high concentrations of helium than do the fcc alloys - the high-temperature ductility is reduced and the fracture mode becomes intergranular. The available neutron results represent very low helium contents - possibly near a threshold; thus, some alloys show reductions in ductility and others do not.

There are no fatigue or fatigue crack growth data for any of the refractory metal or titanium alloys at irradiation damage levels relevant to fusion reactor applications. Data for these alloys in the unirradiated condition are also limited, particularly for the refractory metal alloys. Figure 11 shows crack growth rates at

Fig. 10. Effect of Neutron Irradiation and Injected Helium on the Total Elongation Ratio of 304 Stainless Steel, V-15 Cr-5 Ti, VANSTAR-7, and Mo-0.5 Ti. Compiled from refs. [19], [21-23].

Fig. 11. Fatigue Crack Growth in Annealed 304 Stainless Steel, Annealed Vanadium, and Ti-6 Al-4 V in the Beta Annealed Condition; all Tested at Room Temperature. From refs. [12], [28-30].

Room temperature for annealed 304 stainless steel [12], vanadium [29,30], and Ti-6 Al-4 V in the beta annealed condition [30] using the correlation of crack growth rate (da/dN) and modulus-normalized stress intensity factor ($\Delta K/E$) proposed by Speidel [31]. The increase in crack growth rate for a given ΔK in going from stainless steel to vanadium to Ti-6 Al-4 V is expected. If all other factors are equal (e.g., wall thickness) and if we use the Speidel correlation between (da/dN) and ($\Delta K/E$) to predict crack growth rates, then the higher thermal conductivity, lower thermal expansion and lower modulus of typical vanadium, niobium, and titanium alloys lead to higher heat fluxes to produce a given crack growth rate in these alloys as compared to stainless steel. As shown in Table 5, the difference in heat flux is not directly proportional to the thermal stress coefficient ($\alpha E/K$) due to the (predicted) higher crack growth rates for a given ΔK .

As stated previously there is no fatigue or fatigue crack growth data at radiation damage levels applicable to fusion reactor applications for vanadium, niobium or titanium alloys. For alloys of vanadium and niobium, high concentrations of helium reduce the elevated temperature tensile ductility and cause the fracture mode to change from transgranular to intergranular. When this occurs, the cyclic properties will probably also be degraded.

Table 5. Relative Stress Intensity Factors (ΔK) and heat fluxes (Q) to Give Equal Crack Growth Rates da/dN

A'lloy	E GN/m ²	$\left(\frac{\Delta K_{\text{alloy}}}{\Delta K_{\text{ss}}}\right)$ For Given da/dN	$\left(\frac{\alpha E/K}{\alpha E/K}\right)_{\text{ss}}$ alloy	$\frac{Q_{\text{alloy}}}{Q_{\text{ss}}}$ To produce equal da/dN
304 ss	160	1	1	1
Ti-6Al-4V	72	0.45	0.38	1.2
V-20Ti	110	0.75	0.32	2.4
Nb-1Zr	65	0.41	0.13	3.2

4. SUMMARY

Irradiation of 316 stainless steel and the nickel-base alloy Inconel 600 in a mixed spectrum reactor to produce helium concentrations equal to several tens of MWy/m² exposure with concomitant displacement damage clearly shows that the resultant property changes are more severe than for fast reactor irradiations in which the amount of helium produced is much lower. Fractures remain transgranular at low temperatures (e.g., 350°C) with a transition to intergranular fracture as the temperature is increased. The temperature at which this transition occurs appears to decrease with increasing helium content. Over the entire temperature range the ductilities are lower at high helium concentrations than for low, helium concentrations at equal dpa levels. Above the temperature at which the transition to intergranular fracture occurs, the ductility is reduced to unacceptable levels. No fatigue or fatigue crack growth data are available for material containing high concentrations of helium. Based on trends observed in irradiations to low dpa and helium levels and on tensile behavior at high helium concentrations with displacement damage, large changes in fatigue properties would not be expected at high helium concentrations in the low-temperature transgranular fracture regime. Very large decreases in properties would be expected at high temperatures.

An important observation is that even at high helium concentrations and displacement damage the properties are generally sensitive to composition and structure. Improved performance through alloy development should be possible. Improved ductility and toughness in the low-temperature (transgranular fracture) regime and movement of the trans- to intergranular fracture temperature upwards are reasonable goals.

From the viewpoint of physical properties, the refractory metal alloys of vanadium, niobium, and molybdenum are attractive fusion reactor first-wall materials. No mechanical property data exist at helium concentrations typical of fusion reactor exposure. From irradiations which produced relatively low dpa levels and very low helium concentrations, it appears that

the changes in strength properties with irradiation follow the same general trends with homologous temperature as in the fcc iron and nickel base alloys. At least some of the bcc alloys exhibit two characteristics not found in the fcc systems: The DBTT can be increased above room temperature and possibly into the operating range, and at some temperature and damage levels the uniform elongation is reduced to zero. Recent experiments in which helium was introduced into vanadium and niobium alloys, either with a cyclotron or by the "tritium trick," show that the bcc alloys as a class are not inherently resistant to helium embrittlement — i.e., high-temperature ductility is reduced and the fracture mode is changed from trans- to intergranular. Assuming that the refractory alloys could not be used above approximately $T_m/2$ because of embrittlement, they still offer the potential for higher temperature because of their higher melting points. Very little data exist on fatigue and crack growth even in the unirradiated condition. From the available data in the unirradiated condition and from modulus-normalized stress intensity factor versus crack growth (all at room temperature), the crack growth rates appear to be higher than in the austenitic stainless steels. However, the higher thermal conductivity and lower thermal expansion more than compensate and will allow higher heat fluxes.

5. CONCLUSIONS

There is now sufficient information at helium concentrations and dpa levels appropriate to the fusion reactor to demonstrate that this environment is much more severe than a fast reactor environment from an irradiation damage viewpoint. Properties do, however, remain sensitive to composition and structure and improved alloy performance should be attainable. Materials selection and alloy development must be based on many considerations in addition to radiation effects (e.g., unirradiated physical and mechanical properties, compatibility with breeding and cooling fluids). It is clear from available results that the effects of radiation on properties must be determined at approximately correct helium and dpa levels. How close one must be to the fusion reactor He:dpa ratio to adequately approximate the effects of this environment has not been determined!

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