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The Effects of CTR Irradiation on the Mechanical Properties of Structural Materials

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MASTER

OAK RIDGE NATIONAL LABORATORY

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F. W. Wiffen

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THE EFFECTS OF CTR IRRADIATION ON THE MECHANICAL PROPERTIES OF STRUCTURAL MATERIALS

F. W. Wiffen

ABSTRACT

Mechanical properties of CTR structural materials are important in determining the reliability and economics of fusion power. Furthermore, these properties are significantly affected by the high neutron flux experienced by components in the regions near the plasma of the fusion reactor. In general, irradiation hardens the material and leads to a reduction in ductility. An exception to this is in some complex engineering alloys where either hardening or softening can be observed depending on the alloy and the irradiation conditions. Regardless of this restriction, irradiation usually leads to a reduction in ductility. Available tensile data examined in this paper shows that significant ductility reduction can be found for irradiation conditions typical of CTR operation. Consideration of these effects show that extensive work will be needed to fully establish the in-service properties of CTR structures. This information will be used by designers to develop conditions and design philosophies adapted to avoid the most deleterious conditions and minimize stresses on structures on reactor design. The information will also be used as input to alloy development programs with goals of producing materials more resistant to property degradation during irradiation.

It is clear that a great deal of additional work will be required both to understand the effect of CTR irradiation on properties and to develop optimal alloys for this application.

I. INTRODUCTION

The importance of understanding the effects of irradiation on the mechanical properties of structural materials for CTRs has been well established. A leak in a CTR structure is a shutdown condition and will result in long reactor outages with serious effects on the economics of fusion power. Since high energy neutrons are very penetrating, the effects of radiation will extend well back from the reactor first wall. Irradiation generally also changes the microstructure of the material and

such changes are reflected in the mechanical properties (usually, a degradation results). Designing a CTR reactor requires a knowledge of these properties as a function of reactor service conditions (i.e., stress, temperature, and neutron fluence).

This paper examines the predicted effects of irradiation on the mechanical properties of potential CTR structural components in the high flux region of the reactor. Results of tensile tests will be used to illustrate changes in mechanical properties predicted for CTR operations. While other properties are also important, there are more data on the effect of irradiation on tensile properties. These results can also frequently be used to predict other mechanical properties: for example, there are correlations between tensile properties, thermal creep properties, and fatigue properties. We will also look at the evaluation techniques that are used both to characterize more fully the tensile behavior and to understand the underlying microstructural changes. Such understanding is necessary if we are to extrapolate properties beyond measured conditions and also if we are to select better materials or develop irradiation resistant alloys. These supporting evaluation techniques include transmission electron microscopy, scanning electron microscopy, and optical metallography.

This paper also presents briefly some results on irradiation creep. This property is quite different from the tensile properties, cannot be predicted from tensile properties and is important to the prediction of material response during reactor operations.

Before turning to the discussion of the irradiation effects, we will discuss properties that are of concern and the tests used to evaluate irradiation-produced changes in these, and examine the CTR environment, especially those conditions that relate to the irradiation response. Finally we will look at the ability of fission reactors to simulate fusion reactor conditions. Following this is a discussion of irradiation effects on tensile properties and other property changes.

II. PROPERTIES OF CONCERN AND EVALUATION TESTS USED

There are a large number of properties that can be used in design and can also be used as a measure of irradiation response. One of these is the swelling produced by formation of internal cavities, a subject treated in detail in this workshop by Gerald Kulcinski. It must be kept in mind that this swelling can produce stresses to be borne by the structure.

Table 1 lists several mechanical properties affected by irradiation and limitations these effects may impose on reactor design and performance.

Table 1. Bulk Radiation Effects on Mechanical Properties That May Limit CTR First Wall Lifetime

Property	Limitation
Tensile	Stress Limitation Strain to Failure
Thermal Creep	Deformation Rate Component Time to Failure
Irradiation Creep	Deformation Rate
Fatigue and Creep Fatigue	Strain per Cycle Cycles to Failure
Crack Growth	Strain per Cycle Cycles to Failure

Tensile properties are determined in tests at a constant strain rate, and the load and specimen deformation are measured as a function of time. This gives strength as a function of strain, and also the ductility (amount of deformation to failure). Tensile tests for a fixed irradiation condition are often measured for a range of test parameters such as test temperature or strain rate. As shown in Table 1, tensile results may predict stress or strain limitations on reactor structures.

A second mechanical test commonly used is the creep-rupture test. Creep tests are performed at a constant load, and elongation is measured as a function of time. One reservation about creep measurements is the fact that the postirradiation creep rates are not the same as would occur during reactor operation. Also, the amount of deformation to failure in a postirradiation test is not necessarily the same as deformation occurring in the reactor neutron flux. Since there are few data on postirradiation creep of samples irradiated under conditions that simulate those in a fusion reactor these properties will not be discussed. However, since post-irradiation creep deformation rates are important for design, especially for higher temperatures, these results must be generated at some time in the future. As shown in Table 1, thermal creep (postirradiation creep) can be important to designers in establishing deformation rates for stressed reactor components.

A related but different property is irradiation creep, (i.e., creep while the reactor is operating). This, too, establishes deformation rates for stressed reactor components. The parametric dependence of irradiation creep is quite different than out-of-reactor creep measurements, and it may be possible to obtain higher ductility than observed in postirradiation testing.

Fatigue and creep fatigue properties of irradiated materials are also important in fusion reactor structures, especially for concepts with short burn times where a high number of cycles will occur during the structure lifetime. Fatigue properties may limit the acceptable strain per cycle in such reactors and may also establish lifetime limits based on the number of cycles to failure. There is no information on in-reactor fatigue behavior.

Finally, crack growth can be regarded as an intermediate stage of the previously mentioned failure mechanisms. It is important when leaks, which may occur before complete failure, can limit the system operation. Crack growth rates may limit the acceptable strain per cycle for fusion reactors and, as with fatigue, may limit the cycles to failure.

III. THE FUSION REACTOR ENVIRONMENT

A number of reactor parameters establish the irradiation response of materials. These will be summarized here for completeness. Nonpenetrating fluxes (such as electromagnetic radiation, alpha particles, impurity atoms, etc.) impose temperature gradients on the structural first wall which are important sources of stress but will not affect the bulk properties.

Table 2 lists a number of the parameters of interest in discussing the mechanical properties of a CTR structure. This table includes a listing of some of the potential structural materials, gives typical suggested temperatures and establishes the temperature range of interest over which to evaluate mechanical properties. Thirdly, the table gives the neutron flux and neutronic response for metals in the high flux region. The wall response parameters of interest to this discussion are the production of displacement damage (measured in dpa, displacements per atom), the rate of accumulation of helium produced by (n,α) reactions, and the rate of accumulation of hydrogen produced by (n,p) reactions.

The final parameter necessary to establish the materials response to irradiation is the stress state of the material. Table 3 lists possible stress sources in fusion reactor components. The magnitudes of these stresses have been calculated only in very few cases. The table shows that there are a large number of stress sources. As we look at the mechanical behavior of irradiated metals it becomes clear that any design must achieve low stresses if failure of metal components is to be avoided in reactor operation.

Table 2. Structural Materials and the CTR Environment

1. MATERIALS IN THE HIGH-FLUX REGION
 - Structural Wall - Conventional alloys of Fe-Ni-Cr
Refractory metals V, Nb, Mo
Ceramics SiC, C
 - Shield or Sacrificial Wall - Graphite
Oxides, carbides, nitrides
 - Insulators (Theta Pinch) - Al_2O_3 or better insulator
 - Coolant Channels - Same as structural material
2. TYPICAL SUGGESTED TEMPERATURES
 - Type 316 Stainless Steel - 280-600°C
 - Nb-1% Zr - 420-1050°C
3. NEUTRON FLUX AND WALL RESPONSE
 - Typical Wall Loading 1 MW/m²
 - Flux 3.7×10^{14} n/cm² sec (20% have E > 10 MeV)
 - Wall Response in One Year at 100% Duty Factor

	316 SS	Nb-1% Zr
dpa	10	7.2
He, appm	200	24
H, appm	520	80

Table 3. Stress Sources in Fusion Reactors

1. STATIC STRESSES
 - (a) Mechanical Support
 - (b) Coolant pressures
 - (c) Loading on Ports and Penetrations
 - (d) Magnetic Field Pressures
 - (e) Vacuum Containment Loads
2. SLOWLY CHANGING STRESSES
 - (a) Relaxation of Mechanical Loads
 - (b) Swelling and Swelling Gradients
(Irradiation-Produced cavities)
3. CYCLIC AND START-UP/SHUT-DOWN STRESSES
 - (a) Thermal Gradients
 - (b) Varying Magnetic Fields
 - (c) Differential Thermal Expansion

Adapted from Styris et al.
BNWL- 1961 (1976)

IV. FISSION REACTOR SIMULATION OF THE FUSION REACTOR ENVIRONMENT

Ideally, irradiation of samples to evaluate the effects of a CTR environment would be in a CTR-spectrum neutron source; however, no such neutron source exists. We are left only with facilities that will simulate the CTR neutron spectrum. Of these facilities, only fission reactors have adequate volume to produce enough samples in which to measure the irradiation effects on mechanical properties. It is important to understand the advantages and limitations of these facilities. Since experience has shown that the most valid basis for correlating and comparing the results of irradiation in different facilities is displacement level (dpa) and helium level, it is on these grounds that we examine the fission reactors. Table 4 compares the irradiation response of two metals, type 316 stainless steel and niobium, for projected operation in a fusion reactor operating at 1 MW/m^2 neutronic wall loading and in two high flux fission reactors. The EBR-II reactor is a fast-spectrum reactor and the High Flux Isotope Reactor (HFIR) is a mixed spectrum reactor, with both fast and thermal energy neutrons. Table 4 shows that both fission reactors produce adequate displacement damage to simulate a fusion reactor; however, there are differences in the helium production in these materials. In type 316 stainless steel there is faster than real time helium accumulation in the HFIR reactor, but much lower than real time helium production rates in the fast spectrum EBR-II reactor. Helium production rates in niobium are well below the CTR case for either fission reactors.

The very important implications of this matchup of helium and displacement damage production rates for stainless steel make it worthwhile to identify the source of this helium production. It is the fortunate scheme of nature shown in Fig. 1 that yields this helium. The isotope ^{58}Ni comprises two-thirds of naturally occurring nickel. This reaction sequence is unique to thermal energy neutrons, it does not occur in spectra deficient in thermal neutrons as that of EBR-II or at the CTR first wall. The other product of this reaction, iron, is generally a constituent of conventional alloys; thus it has no deleterious effect on the composition of the alloy.

Table 4. Damage Parameters for CTR First-Wall and High-Flux Fission Reactors

Facility	Displacements (dpa/yr)	Helium (appm/yr)
<i>Type 316 Stainless Steel</i>		
CTR ^a	10	200
EBR-II	35	5
HFIR	35	1900
<i>Niobium</i>		
CTR ^a	7.2	25
EBR-II	23	1
HFIR	20	2

^aCTR at 1 MW/m²

Fig. 1 Thermal Neutron Capture Sequence in Nickel That Yields High Helium Production Rates.

In fission reactor irradiations it is relatively easy to control the temperature, stress state and the chemical environment. Mechanical property samples can be large enough for accurate measurements and usually enough samples can be accommodated for full test matrices. EBR-II and HFIR irradiation experiments do not have active temperature control. Irradiation temperatures above the reactor coolant temperature are achieved in experimental capsules that use the sample nuclear heating. The design shown in Fig. 2 has been used to irradiate samples in EBR-II at temperatures up to 1500°C. Various passive temperature sentinels evaluated after the irradiation establish the actual irradiation temperature. Figure 3 shows the type of experiment used to produce elevated temperature irradiation in the HFIR reactor. As the nuclear heating rate in HFIR is much higher than in EBR-II smaller samples and much more careful control of heat flow are required. In both designs the heat generated in the samples is transferred across a gas gap to establish the temperature gradient giving the desired sample irradiation temperature. The samples shown in Fig. 2 and 3 can be used for tensile and creep property measurements. The same samples could be used for fatigue studies if the fatigue cycle is limited to tensile loading. Pressurized tubes can be irradiated in a similar configuration and other specimen geometries are possible.

Fig. 2. Schematic of Tensile Specimens and Specimen Holder Used in EBR-II Materials Irradiation Experiments.

Fig. 3. Schematic View of HFIR Materials Irradiation Experiment. This experiment achieves irradiation temperatures well above the reactor coolant temperatures.

V. IRRADIATION EFFECTS ON TENSILE PROPERTIES

Neutron irradiation of metals produces changes in the microstructures which change the strength. Commonly, hardening and a loss in ductility result, although in some cases softening is observed. While these are not totally separable phenomenon they will be discussed separately in this section.

A. Changes in Strength Properties

Irradiation effects have been frequently described as "radiation hardening" because irradiation generally strengthens materials. An example of this hardening, determined by increase in the ultimate tensile stress, is shown in Fig. 4. These results are for molybdenum irradiated to high fluences at two irradiation temperatures. The yield strength and ultimate tensile strength were increased by a factor of 2 to 3 relative to that of unirradiated material. For a fixed irradiation

Fig. 4. The Ultimate Tensile Strength for Irradiated and Control Samples of Molybdenum. Tensile strain rate was 0.02 min^{-1} except as shown. Solid lines connect results differing only in test temperature. Dashed lines connect results where irradiation conditions or strain rates are not held constant.

condition, the strength values decreased with increasing test temperatures. This strengthening results from the fine-scale defects in the microstructure. Figure 5 shows the microstructures that accompany these property changes. The sample in Fig. 5(a) was irradiated at 455°C , producing a very high concentration of small, unresolved dislocation loops. Irradiation at the higher temperature produced a much lower concentration of dislocation lines, rather than the compact loops shown in Fig. 5(a). However, the main microstructural feature is the high concentration of small cavities shown in Fig. 5 (b). It is these microstructural features, dislocation lines, small dislocation loops, or cavities, which impede the motion of deformation dislocations and produce the strengthening shown in Fig. 4.

The increase in strength has been successfully correlated with the microstructural observations in a number of cases. For small dislocation loops or cavities the strengthening is given by:

Fig. 5. Microstructures in Molybdenum. (a) Irradiation to 3.5×10^{22} n/cm² at 450°C produces a high concentration of dislocation loops and fewer voids. (b) Irradiation to 6.1×10^{22} n/cm² at 857 to 1136°C produced many voids but a very low concentration of dislocation structure.

$$\Delta\sigma_i = \alpha_i Gb \left[\sum_j C_j d_j \right]^{1/2},$$

where: $\Delta\sigma_i$ is the strengthening produced by the i defect type

α_i is a constant for the i defect type

G is the shear modulus

b is the Burgers vector of the slip dislocation

C_j is the concentration of i defects in the size range d_j

d_j is the diameter of the defects.

The summation is over all sizes of the i defects. For dislocation lines, a similar expression is:

$$\Delta\sigma_\rho = \alpha_\rho Gb(\rho)^{1/2},$$

where ρ is the dislocation density measured as the number of lines crossing unit area.

If these strength increases were the only effects of irradiation, irradiation would be beneficial, since a stronger structure can support

higher loads. However, the hardening generally is accompanied by a loss of ductility and this will be treated in detail later.

Irradiation does not always harden materials. In some of the more complex engineering alloys, e.g., stainless steel, strengthening of the unirradiated alloy may be achieved by a controlled distribution of second phases, precipitates, or by cold working to introduce a high concentration of dislocations. Irradiation-produced changes can interfere with these microstructures. On irradiation, additional precipitation may occur, existing precipitates may coarsen, and cold worked structures may recover. Figure 6 shows an example of strength decreases during irradiation of a highly strengthened stainless steel. Over a wide range of irradiation and test temperatures, the irradiation has reduced both the yield and ultimate tensile strengths relative to the unirradiated material. The main microstructural change that produces this strength loss is a decrease in the dislocation concentration. Figure 7 shows the microstructure in a

Fig. 6. The Strength Values 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 He in the samples.

Fig. 7. Type 316 Stainless Steel, 20% Cold Worked, Irradiated at 450°C to 54 dpa and 3660 appm He.

cold worked sample irradiated at 450°C. This microstructure contains a high concentration of dislocations (but fewer dislocations than before irradiation), a high concentration of small cavities (the white spots) and large amounts of precipitate phase.

Having now examined briefly the changes in strength properties produced by irradiation we will turn to the other tensile properties of major interest, the ductility.

B. Ductility and Failure Modes

Irradiation reduces ductility through a number of processes. These will be separated into four distinguishable modes for discussion in this section.

1. Ductility Loss by Hardening — There is a generally observed inverse relationship between hardening and ductility in a metal. Most processes that harden a material also reduce its ductility. This inverse relationship for irradiation is illustrated in Fig. 8. These tensile curves for niobium show the yield strength and ultimate tensile strength increased by irradiation. At the same time, the ductility has been reduced. Both the uniform elongation (the elongation to the point of maximum load) and the total elongation (the elongation to failure) are reduced by the irradiation. This hardening was explained earlier. The ductility loss by hardening is in the uniform elongation, reduced in this case by a factor of about 4. The reduction in uniform elongation results from an increase in the strain hardening rate, which reduces the strain hardening exponent. As a first approximation the strain hardening exponent and uniform elongation are equal and thus the uniform elongation is reduced on hardening. Ductility loss by this mechanism is commonly observed and results in a reduced elongation in a tensile test.

Fig. 8. Ductility Loss by Hardening in Neutron Irradiated Niobium. Irradiation in EBR-II to 3×10^{22} n/cm² at 460°C.

2. Embrittlement by DBTT Shift — A number of materials are sensitive to failure in a cleavage mode with almost zero plastic deformation. This failure mode is very sensitive to material and test conditions.

For a given material, and fixed deformation rate, there exists a temperature below which the material is fully brittle and above which there is ductile deformation. This temperature is called the ductile-to-brittle transition temperature (DBTT). This mechanism of embrittlement is important in the ferritic steels used in fission reactor pressure vessels. Of the potential CTR structural materials, molybdenum alloys are the most sensitive to this mechanism. Figure 9 shows the type of tensile curves observed. In this case an irradiated molybdenum alloy deformed only along the elastic modulus line, failing with no plastic or permanent elongation.

The DBTT behavior can be explained at least qualitatively on the basis of the Ludwig-Davidenkov diagram, a modified form of which is shown in Fig. 10. In this figure a band defines the cleavage fracture stress and tests in which the stress reach this range fail in a cleavage fracture mode. Four tensile tests are plotted in Fig. 10, superimposed on the cleavage stress range. In the first two tests the specimens were loaded into the cleavage fracture stress range then failed abruptly, by a

Fig. 9. Brittle Fracture Produced in Mo-0.5% Ti by Neutron Irradiation. Irradiation in EBR-II to 3×10^{22} n/cm² at 425°C.

Fig. 10. Tensile Curves for Molybdenum Irradiated to $3.5-4.0 \times 10^{22}$ n/cm² at 455°C. The tensile test conditions are shown. These tensile curves are superimposed on the cleavage stress band to demonstrate the source of embrittlement.

cleavage mode with low elongation. In the third test the material deformed, the stress stayed below the cleavage stress, the specimen exhibited over 6% elongation and failure occurred in a ductile mode. The fourth test shown in Fig. 10 was unique. The material loaded along the elastic modulus line, then deformed gradually, hardening as further deformation was obtained. When the stress reached the cleavage fracture range failure occurred abruptly by a cleavage mode.

The cleavage fracture mode, characteristic of DBTT failures, can be established by examining the fracture surface. Figure 11 shows fracture surfaces examined by the scanning electron microscope of Mo-0.5% Ti samples tested under different conditions. Figure 11(a) shows the characteristic cleavage fracture, with the fracture propagated through the material without any secondary cracking and with little deformation in the fracture zone. Figure 11(b) shows failure in a ductile test, with the characteristic dimples indicating a high amount of local deformation.

Fig. 11. Typical Areas on the Fracture Surfaces of Irradiated Mo-0.5% Ti [3.0×10^{22} n/cm² at 425°C] Tensile Specimens. Tests at 400°C, (a) at strain rate of 0.02 min⁻¹. (b) at 0.0002 min⁻¹. Scanning electron microscope magnification 1000×.

The most important test variable for materials susceptible to cleavage fracture is the deformation rate. In general the DBTT shifts upward in temperature as the strain rate is increased. Alternatively, at a fixed temperature the observed behavior can depend on the strain rate. Tests 2 and 3 in Figure 10 show that for fixed irradiation conditions and a constant test temperature a high strain rate resulted in cleavage fracture while a low strain rate produced a ductile-mode failure.

The fracture mechanism at temperatures below the DBTT is independent of irradiation. The effect of irradiation is to harden the material so that it reaches the cleavage fracture stress range. The cleavage fracture stress is independent of test temperature and independent of irradiation. Fracture surfaces for cleavage failures are not distinguishably different regardless of pretest and test conditions.

3. Ductility Limited by Plastic Instability — The third mechanism of ductility loss observed in irradiated materials is due to plastic instability. Characteristics of this mechanism are shown in the two tensile curves in Fig. 12. The control sample tensile response is deformation with increasing stress until an ultimate load is reached and

Fig. 12. Ductility Loss Due to Plastic Instability. This example is in irradiated Mo-0.5% Ti. Irradiation in EBR-II to 3×10^{22} n/cm² at 425°C.

then a gradual decrease in load on further deformation until failure occurs. The irradiated sample deformed to a maximum load at almost zero plastic deformation. Further elongation occurred with continually decreasing strength to failure. The uniform elongation (i.e., the elongation to maximum load) is near zero for this type of deformation. Once the material yields, its load carrying ability continually decreases with deformation.

This type of ductility loss has been observed in several bcc materials irradiated at relatively low temperatures. The type of microstructure observed in these cases is shown in Fig. 5(a) and Fig. 13. The dominant defect structure is a high concentration of dislocation loops. The deformation mechanism in samples with this microstructure is shown in Fig. 14. Deformation caused the narrow white channels seen in Fig. 14(a). These channels are cleared as deformation dislocations move through the microstructure, removing the hardening dislocation loops. This weakens the channels for further deformation on the same or closely parallel planes. In Fig. 14(b) the traces observed are traces of {110} planes,

Fig. 13. Microstructure of Nb-1% Zr Irradiated to 3.7×10^{22} n/cm² at 448°C.

Fig. 14. Channeling Deformation in Neutron Irradiated Tantalum. This microstructure found in a tantalum sample irradiated to 2.5×10^{22} n/cm² at 425°C and deformed during thinning. The clean channels are swept free of the dislocation loops (black spots) by glide dislocations.

the expected slip planes for a bcc material. With continuing deformation the load required to drive the deformation decreases. Dislocation loops, and noncompact defect clusters, are susceptible to this type of deformation. On the other hand, voids or cavities are resistant to this type of deformation, and cases have been seen where the dislocation loops are removed by deformation but although voids are sheared by the deformation they are not removed.

The effect of irradiation temperature, which translates to the effect of microstructure, is shown in Fig. 15, comparing tensile tests in tantalum samples irradiated at two temperatures but tested at a common temperature. By inference, we know that the lower temperature irradiation produced a high concentration of small loops, but no voids, and deformation in the channeling mode leads to zero uniform strain. Irradiation at the higher temperature produces a microstructure containing fewer dislocation loops but a high concentration of voids. These voids interact with deformation dislocations to work harden the material and allowed uniform elongation of 5 to 6%. This suggests that there may be cases

Fig. 15. The Effect of Irradiation Temperature, and Thus of Microstructure, on Plastic Instability in Tantalum. Irradiation in EBR-II was to comparable fluences and tensile tests were run at 400°C.

where reactor operation at higher temperatures would impart an advantage relative to lower temperature operation. It also suggests that there may be cases where it is an advantage to promote the formation of cavities, whether they be voids or helium bubbles, to avoid a microstructure dominated by only dislocation loops and thus susceptible to plastic instability.

Figure 15 in itself might suggest that plastic instability is solely a function of the amount of hardening produced by the irradiation. This was ruled out by an experiment wherein niobium samples were irradiated at two different temperatures and tested at a common temperature. The results are shown in Fig. 16. The niobium irradiated at 55°C contains a high concentration of dislocation loops and no voids, the material deformed by dislocation channeling with the maximum load at yield. The sample irradiated at 460°C contains a microstructure which is dominated by the void concentration (a high void concentration and a lower loop concentration). This sample yielded at a strength well above the strength of the sample irradiated at 55°C but work hardened after yielding strength of the sample irradiated at 55°C but work hardened after yielding

Fig. 16. The Dependence of Plastic Instability on Microstructure. This shows that in niobium, for fixed tensile test conditions, plastic instability is not solely a product of the tensile strength achieved but depends on the type of microstructural feature that produces the hardening.

to an ultimate load at 11% uniform elongation. This shows that plastic instability can occur at a lower stress than the stress at which some uniform elongation is retained; the difference is in the microstructural response of the material. The void-dominated microstructure has a higher ductility than does a loop-dominated microstructure.

This plastic instability, due to channeling, is generally of less importance in stainless steel than in bcc metals. While channel deformation has been observed in stainless steel after high fluence irradiation, the channeling has usually occurred only after some uniform elongation.

4. Helium Embrittlement — This ductility loss mechanism will be the most serious concern to the success of CTR operation. The helium is produced by (n,α) reactions; and in stainless steel the rate of accumulation is ~ 200 appm/MW-yr/m² of reactor operation. The helium promotes intergranular fracture which is a low ductility fracture mode. The helium migrates to the grain boundaries, and reduces the fracture stress. The deleterious effect is magnified by higher temperature service, and can occur independent of, or be enhanced by, the lattice hardening produced by displacement damage.

An extreme example of helium embrittlement in Inconel 600 is shown in Fig. 17. Irradiation at 700°C produced 7 dpa and 1260 appm He. The helium greatly reduced the fracture strength of the material, with fracture at zero elongation.

The dependence of helium embrittlement on irradiation temperature is illustrated in Fig. 18. At low temperatures, low but measurable ductilities were achieved. With increasing test temperature, the ductility was reduced until at 650°C both cold worked and annealed material showed ductility $< 0.5\%$. The fracture mode of helium embrittlement is grain boundary separation. Helium embrittlement and cleavage fractures, both of which can have zero elongation, can be distinguished by the appearance of the fracture surface. Cleavage fractures propagate through grains but helium embrittlement follows grain boundaries. Figure 19 shows a cross section through a stainless steel sample that had failed with essentially zero ductility at 650°C. The fracture path follows the grain

Fig. 17. Helium Embrittlement in Inconel 600 Irradiated in the HFIR Reactor.

Fig. 18. Tensile Curves for Type 316 Stainless Steel Irradiated in HFIR. Irradiation at temperatures slightly above the test temperature produced 40 to 60 dpa and 3000 to 4200 appm He.

Y-127365

Fig. 19. Photomicrographs of Fracture Region of Solution Annealed Type 316 Stainless Steel Irradiated at 685°C in HFIR and Tensile Tested at 650°C. (a) Unetched and (b) etched.

boundaries without appreciable secondary cracking. The first crack formed has propagated across the sample without branching. Scanning electron microscopy examination of a fracture is shown in Fig. 20. The fracture surface is composed of well-defined, relatively clean grain boundaries where separation has occurred. Some evidence of cavities that have been formed by coalescence of helium on these boundaries can be seen. Comparison of this figure with Fig. 11 shows the difference between the well-defined grain boundaries of helium embrittlement, the cleavage marked fracture surface of DBTT brittle fracture, and the dimples of ductile fracture.

Microstructural examination of samples with high helium concentration reveals the features that are characteristic and suggests the reason for the grain boundary fracture. Figure 21 shows the microstructure of stainless steel irradiated to produce almost 1800 appm He at 574°C. The feature of note in this microstructure is the almost continuous line of large cavities along the grain boundary. This path is followed by the fracture.

Fig. 20. Scanning Electron Microscope Image of Tensile Fracture Surface of Inconel 600 Specimen Tested at 700°C. HFIR irradiation of this sample at 700°C produced 7 dpa and 1250 appm He. Fracture was by grain boundary separation.

Fig. 21. Microstructure of a Type 316 Stainless Steel Irradiated in the Solution Annealed Condition in HFIR. Irradiation at 574°C produced 29 dpa and 1790 appm He. The nearly continuous line of helium bubbles in the grain boundary leads to greatly reduced tensile elongation in tests of similar specimens.

Fig. 23. Creep of Pressurized Tubes of 20% Cold Worked Type 316 Stainless Steel. Lines labeled with test times are for unirradiated tests, data points and lines labeled with displacement level are for pressurized tubes irradiated in the EBR-II. The times of control tests correspond approximately to the time at temperature of the irradiated specimens. Figure supplied by E. R. Gilbert, Westinghouse Hanford Company.

Fig. 24. Swelling and Irradiation Creep Results from the Creep-I Experiment. Results from periodic profilometry of pressurized tubes irradiated in EBR-II at 415°C. From G. L. McVay, *Trans. Am. Nucl. Soc.* 23, 147 (1976).

Helium embrittlement is of course a function of the helium level. Figure 22 shows the dependence of tensile ductility at 575°C on neutron fluence and associated helium concentration. At this test temperature the total elongation drops rapidly from the range 6 to 9% in unirradiated samples to 0.5 to 3% for helium levels less than 100 appm. Further irradiation to high helium levels produces some reduction in sample elongation but without a drop to zero elongation.

The final comment to be made on helium embrittlement is that there has been no complete solution identified for the problem. Some preliminary

Fig. 22. The Tensile Ductility of 20% Cold Worked Type 316 Stainless Steel Irradiated in the HFIR. Helium levels produced by the irradiation are shown on the data points, and compare to ~ 200 appm He produced per year of CTR operation at a wall loading of 1 MW/m^2 .

results suggest that some alloys may be tailored to improve the resistance to helium embrittlement, but it is unlikely they will be totally immune.

C. Tensile Properties Summary

The effect of irradiation on tensile properties has been reviewed in earlier parts of this chapter. We have seen that irradiation generally produces an increase in the strength properties but may in some cases produce a loss of strength. Of greater importance is the loss in ductility produced by the several mechanisms reviewed above. Table 5 summarizes the observed ductility loss mechanisms, gives examples of materials in which

Table 5. Ductility Loss Mechanisms

(1) He Embrittlement
SS, Ni Alloys, Al Alloys
High Temperature Application
(2) DBTT
Mo Alloys
Coolant Temperatures
(3) Plastic Instability (Channeling)
Nb, Mo, SS
(4) Hardening
Nb, Al, (SS)

these mechanisms have been observed, and gives some indication of the temperature range over which the mechanisms is important. A summary in Table 6 gives the mechanism limiting ductility for several suggested CTR alloys systems. The most severe restraint and secondary restraint are listed. It should be emphasized that this summary is based on scanty data and a large measure of personal judgment.

Table 6. Mechanisms Limiting Ductility of Irradiated CTR Materials

Alloy System	Most Severe Restraint	Secondary Restraint
Stainless Steel and Nickel Base	He Embrittlement	Matrix Hardening
Aluminum Base	He Embrittlement	Hardening
Mo or W Base	DBTT Shift	Plastic Instability
Nb or Ta Base	Plastic Instability	Hardening
V Base	Hardening	He Embrittlement

Reservation: This summary developed from a very limited data base.

VI. OTHER POSTIRRADIATION MECHANICAL PROPERTIES

Although tensile property data are most abundant there are a number of other properties that could be used to measure irradiation effects and may be important in reactor design. Table 7 gives a list of properties

Table 7. Mechanical Properties After Irradiation

Causes:

Microstructural changes control matrix strength
 Insoluble gases affect fracture mode
 Several processes reduce ductility:
 plastic instability, DBTT shift,
 reduced fracture strength

Critical Parameters:

Fluence — dpa and gas production
 Temperature of irradiation
 Composition
 Preirradiation microstructure

Measurements:

Tensile, creep, or fatigue test
 Strength values
 Ductility values
 Microstructure and fractography

and considerations that relate to all postirradiation mechanical properties. The critical parameters controlling the observed property changes are the fluence, as measured both by displacements per atom and by gas production, and the irradiation temperatures. The results of tensile tests, postirradiation creep rupture tests, fatigue tests, or creep fatigue interaction tests can all be important to designers. Somewhat different types of tests are used to determine crack growth rates and these will also be important. Table 7 also suggests measurements that would be used in evaluating postirradiation mechanical properties.

The emphasis on tensile properties in available irradiation effects data reflects mainly the need of fission reactor programs. As programs focus more intensively on the needs of the CTR programs, greater emphasis will be placed on fatigue testing and other tests judged to be of prime importance. Since little data are available on the effect of irradiation on fatigue these considerations can not be discussed in this paper.

VII. IN SITU PROPERTY MEASUREMENTS

Some irradiation effects are the result of the flux of point defects rather than of the microstructure that results from the precipitation of these point defects. When this is the case *in situ* measurements of the properties are required to evaluate the irradiation effect. Of great interest is irradiation creep; the deformation rate of a stressed material can be controlled by the biased flow of point defects to dislocations. Complete measurement of irradiation creep rates require sophisticated experiments, with measurements performed during irradiation. As a result of this difficulty, the amount of data available is severely limited, but the field of irradiation creep is receiving much attention and a great deal of information will be generated in the next few years.

Some of the considerations that relate to irradiation creep are given in Table 8. The problem is being attacked both in very sophisticated in-reactor experiments with deformation rates measured during irradiation and more simply in experiments in which pressurized tubes are irradiated and periodically withdrawn from the reactor for tube diameter measurements.

Table 8. Irradiation Creep

<u>Cause:</u>	Biased flow of point defects control deformation Deformation rates exceed out-of-reactor rates
<u>Critical Parameters:</u>	Flux — dpa generation rate
<u>Modifying Parameters:</u>	Fluence Stress Temperature Composition and metallographic state
<u>Measurements:</u>	Stress, temperature Strain with high precision

Comparison with out-of-reactor creep rates then establishes the dependence of the deformation rate on the reactor parameters.

An independent way in which irradiation creep parameters are being determined is in simulation with high energy light ion bombardment of thin samples. The deformation rates measured on these small samples are then related to the displacement rates during irradiation.

A few irradiation creep data are now available. Figure 23 shows a comparison of deformation in pressurized tubes of 20% cold worked type 316 stainless steel out-of-reactor and during reactor irradiation. The amount of strain at low temperatures is much greater during irradiation than for an equivalent time at the same stress and temperature out-of-reactor. For type 316 stainless steel at the displacement rates of EBR-II, irradiation creep is the dominate creep mechanism for temperatures below about 600°C. Irradiation creep is much less temperature dependent than is thermal creep, and at lower temperatures strain is approximately linear with dpa, i.e. with irradiation time. It has also been suggested that the strain is linear with stress as well. Figure 24 shows similar results for type 304 stainless steel. The diameter increase in the unpressurized tube is due to swelling of the material. When this is

accounted for in the pressurized tubes, the remaining irradiation creep is approximately linear with fluence and with stress level.

VIII. SUMMARY

Mechanical properties of CTR structural materials are important in determining the reliability and economics of fusion power. Furthermore, these properties are significantly affected by the high neutron flux experienced by components in the regions near the plasma of the fusion reactor. In general, irradiation hardens the material and leads to a reduction in ductility. An exception to this is in some complex engineering alloys where either hardening or softening can be observed depending on the alloy and the irradiation conditions. Regardless of this restriction, irradiation leads to a reduction in ductility. Available tensile data examined in this paper show that significant ductility reduction can be found for irradiation conditions typical of CTR operation. Consideration of these effects show that extensive work will be needed to fully establish the in-service properties of CTR structures. This information will be used by designers to develop conditions and design philosophies adapted to avoiding the most deleterious conditions and minimizing stresses on reactor structures. The information will also be used as input to alloy development programs with goals of producing materials more resistant to property degradation during irradiation.

It is clear that a great deal of additional work will be required both to understand the effects of CTR irradiation on properties and to develop optimal alloys for this application.

IX. ADDITIONAL READING LIST

While this paper has not referenced original data, there are a number of comparable surveys of irradiation effects expected in fusion reactors and on the use of the available data in designing fusion reactors. These articles closely parallel the present report and reference is made to the original literature. Several articles bearing on this subject are listed below:

(A) General Discussion of Radiation Effects in the CTR Context.

1. F. W. Wiffen, "Radiation Damage in CTRs," p. 140 in *Proceedings of the International Working Sessions on Fusion Reactors Technology*, USAEC Report CONF-710624 (1972).
2. L. C. Ianniello, ed., *Fusion Reactor First-Wall Materials*, USAEC Report WASH-1206 (1972).
3. G. L. Kulcinski, compiler, "Bulk Radiation Damage to First-Wall Materials," p. 479 in *Fusion Reactor Design Problems*, IAEA (1974).
4. F. W. Wiffen, "Radiation Effects in Structural Materials for Fusion Reactors," to be published as chapter 5 of *Critical Materials Problems in Energy Production*, Academic Press (1976).

(B) Discussion of Limitations Imposed on CTR Design by Radiation Effects to Metal Structural Components.

5. G. L. Kulcinski, R. G. Brown, R. G. Lott, and P. A. Sanger, "Radiation Damage Limitations in the Design of the Wisconsin Tokamak Fusion Reactor," *Nucl. Technol.* 22, 20 (1974).
6. E. E. Bloom, F. W. Wiffen, P. J. Maziasz and J. O. Stiegler, "Temperature and Fluence Limits for a Type 316 Stainless Steel CTR First Wall," to be published in *Nuclear Technology* (1976).

(C) Conceptual Reactor Designs Which Consider Radiation Effects Limitations.

7. B. Badger et al., "UWMAK-I, A Wisconsin Toroidal Fusion Reactor Design," UWFD-68, University of Wisconsin, November 1973.
8. A. P. Fraas, "Conceptual Design of the Blanket and Shield Region and Related Systems for a Full-Scale Toroidal Fusion Reactor," ORNL-TM-3096 (May 1973).
9. R. G. Mills, ed., "A Fusion Power Plant," Plasma Physics Laboratory, Princeton University Report MATT-1050 (August 1974).

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