DOE/ER-0046/13
Vol. 2

Damage Analysis and Fundamental Studies

Proceedings of the Second Workshop on
Fusion Environment Sensitive Flow and Fracture Processes

May 1983

U.S. Department of Energy
Office of Energy Research
Office of Fusion Energy
Washington, DC 20545

Printed in the United States of America
Available from
National Technical information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

NTIS price codes
Printed copy: A 04
Microfiche copy: A01

DOE/ER-0046/13
UC-20,20c
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FUSION ENVIRONMENT SENSITIVE FLOW AND FRACTURE PROCESSES

A workshop held on
August 2-3, 1982
Pacific Northwest Laboratory
Richland, Washington

Organized by:

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Damage Analysis and Fundamental
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CONTENTS

I. SUMMARY

The second working meeting on "Fusion Environment Sensitive Flow and Fracture Processes was held for the purpose of reviewing the progress made since the last meeting on this topic held in August 1980. The workshop report on the 1980 meeting (DOE/ER-0046/4, Vol. 2) was used as a guide for assessing accomplishments in the last two years and for updating the research recommendations. The approach used for the second working meeting differed slightly from the first in that the discussion leaders divided their time between status reports and discussions on the recommendations. The participants were not divided into groups but remained together to ensure that 'input on the recommendations was received from the group as a whole.

The recommended research activities are summarized below in order of decreasing priority within each topical area. Priorities are relative to activities within each working group, as no overall priorities were established. The priorities for these activities reflect either the near term need for the research or the lack of information on a particular property. It is apparent that the effort required to satisfactorily complete the list of recommended research activities exceeds the present program funding; however, this list is expected to serve as a guideline for new programs and existing programs. Also, it is recognized that many of these activities are supportive of the Alloy Development for Irradiation Performance (ADIP) goals, and as such a close liaison with ADIP must be maintained. It is expected that activities supportive of ADIP goals will be performed in such a way that the results will be of assistance in alloy development and selection. Discussion of the recommended research can be found in Section IV of this report.

RESEARCH RECOMMENDATIONS: SUMMARY

A. FLOW PROCESSES AND PROPERTIES: N. M. Ghoniem, Discussion Leader

1. Assess effects of helium on irradiaton creep - high priority.

2. Develop irradiation hardening - enibrittlement relationships for ferritic steels - high priority.

3. Further modeling and evaluation of creep-rupture - high priority. (Also discussed and recommended in Time Dependent Fracture section)

4. Further modeling and evaluation of pulsing effects - medium priority.

5. Further evaluation of cascade effects on irradiation creep - low priority.

B. EFFECT OF FLOW PROCESSES ON FRACTURE: W. G. WOLFEH, Discussion Leader

1. J_{IC} measurements and fracture mode determination of type 316SS irradiated to high doses - high priority.

2. Experimental and analytical evaluation of the transgranular to intergranular transition temperature in irradiated 316SS - high priority.

3. Evaluate effect of fusion environments on ΔK_0 and first wall lifetimes - high priority.

4. Evaluate the fracture properties of welds and heat affected zones in irradiated 20% CW 316SS - low priority.

5. Analyze the effect of the softening observed in irradiated 20% CW 316SS on fracture processes - low priority.

C. TIME DEPENDENT FRACTURE: G. Lucas, Discussion Leader

1. Complete and circulate a topical report and recommendations on fatigue crack growth by June 1983 - high priority.

2. Continue development of calibrated correlation methodologies for creep-rupture - high priority.

3. Summarize Breeder creep-rupture data in a topical report - high priority.

4. Flag ADIP creep-rupture specimens for microstructure archives - high priority.

5. Assess alternate pressurized tube designs for evaluating stress state effects - medium priority.

6. Continue monitoring literature on creep crack growth and write a summary report by Aug. 1984 - medium priority.

7. Evaluate the effect of strain rate on irradiation embrittlement of .austenitic stainless steel - medium priority.

8. Evaluate fusion relevant creep-fatigue interactions - medium priority.

D. KADIATIUN EMBKITTLEMENT AND ENVIRONMENTAL EFFECTS: R. H. Jones, Discussion Leader

1. Develop micro-hardness/yield strenyth correlations for irradiated 316SS and HT-9 - hiyh priority.

2. Continue development of model based and experimental property - property correlations for fracture and fatigue crack yrowth - high priority.

3. Continue evaluation of effects of irradiation on impurity segreyation - high priority.

4. Evaluate the effect of grain boundary phosphorus segregation on the stress corrosion of 316SS - high priority.

5. Evaluate and model the relationship between segregation, precipitation, irradiation hardening, and fracture processes of ferritic alloys - high priority.

6. Assess the status and relevance of stress corrosion and corrosion fatigue models for fusion environments and publish a status report by August 1984 - medium priority.

7. Continue development of techniques for evaluatiny yrain boundary chemistry of irradiated materials - low priority.

II. WORKSHOP SCHEDULE

Monday, August 2

A. Radiation Embrittlement and Environmental Effects - R. H. Jones, Discussion Leader

8:30 a.m.	Status Report
10:00	Break
10:15	Discussion of Recommendations/Priorities
12:00	Lunch

B. Flow Processes - N. Ghoniem, Discussion Leader

1:30 p.m.	Status Report
3:00	Break
3:15	Discussion of Recommendations/Priorities
5:00	Adjourn for dinner.

Tuesday, August 3

C. Effects of Flow Processes on Fracture - W. G. Wolfer, Discussion Leader

8:30 a.m.	Status Report
10:00	Break
10:15	Discussion of Recommendations/Priorities
12:00	Lunch

0. Time Dependent Fracture - G. E. Lucas, Discussion Leader

1:30 p.m.	Status Report
3:00	Break
3:15	Discussion of Recommendations/Priorities
5:00	Adjourn

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IV. RECOMMENDATIONS: STATUS AND UPDATE

A. FLOW PROCESSES AND PROPERTIES Discussion Leader: N. M. Ghoniem, University of California, Los Angeles.

1. Overview and Status Report

The session on flow processes and properties was organized into presentations and discussions. Presentations were grouped together for coherent and balanced understanding of the issues involved. The following presentations were given:

1. Status of Research since August 1980 (Ghoniem)
2. Accomplishments and progress (Ghoniem)
3. Influence of flow processes on design (Ghoniem)
4. Cyclic flux effects on irradiation creep (Ghoniem)
5. Theoretical relationships between swelling and creep during irradiation (Mansur)
6. Radiation hardening at low temperature (Ghoniem)
7. Radiation hardening at high temperature (Garner)
8. Hardness-strength correlations (Panayotou).

The discussions on flow processes and properties focused on irradiation-induced swelling, creep, and hardening. It was first felt that properties should be studied in perspective with fusion reactor designs. In order to insure that experimental programs and theoretical efforts lead to desirable design criteria, the implications of changes in properties due to irradiation were discussed. The microstructural origins of flow properties, as well as property-property correlations received considerable attention.

The area of irradiation effects on creep properties has been the center of extensive discussions. At temperatures below ~400-450°C in steels, irradiation creep plays a significant role as a deformation mechanism for reactor structural components. It has been recognized that creep deformation has to be considered in relationship to swelling deformation in order to translate these two processes to design failure criteria. Conceptual reactor designs thus far have employed crude definitions of component failure mechanisms. Residual stresses generated by differential deformation rates may lead to an undesirably high stress that may cause failure. The complex geometries of some fusion blanket structures may not tolerate excessive gross structural deformation. The absolute magnitudes of swelling and creep and their ratio are important factors for failure analysis. Irradiation creep mechanisms were reviewed, with emphasis on a newly proposed model (1,2) of cascade-induced dislocation climb.

At temperatures above approximately 450°C in steels, failure mechanisms due to creep deformation were more clearly defined. The slow creep-rupture process is recognized to be the cause of failure. Considerable uncertainties appeared to exist, however, regarding the effects of irradiation on classical creep-rupture processes. The limited available in-reactor data show either an improvement or a decrease in the time-to-rupture due to irradiation. Post-irradiation creep measurements show a consistent decrease in the rupture time due to irradiation. Fundamental understanding of the issues governing these processes is sorely needed. This is particularly true when reactor structural components are under the influence of irradiation during only limited fractions of their operational lifetime. Such a situation may arise due to startup/shutdown procedures, scheduled maintenance, or the inherent pulsed nature of some fusion devices.

Irradiation hardening phenomena were then discussed. It was felt that the mechanisms of irradiation hardening are better understood and documented than two years ago (3). The influence of irradiation on increasing the tensile strength of structural alloys can be broadly classified into low-temperature, and high-temperature mechanisms. Low-temperature hardening was identified with the formation of defect clusters, both vacancy and interstitial loops. On the other hand, the origins of high-temperature hardening were analyzed and related to various microstructural features.

There was an overall feeling of progress in understanding flow processes over the last two years. However, uncertainties were perceived to exist in certain areas. A sense of the importance of linking the work on properties to design issues prevailed. Our recommendations were therefore slanted toward a direction that is more meaningful to designers, but yet fundamentally based. Table 1 summarizes the achievements, or lack of progress, in the previously recommended research areas since August 1980 (3). This is followed by a section on the current recommendations of the working group.

TABLE 1

PROGRESS IN FLOW PROCESSES AND PROPERTIES

	Research Area	Progress
1.	Correlation of Radiation Hardening with Microstructure (Low Fluence)	Significant
2.	Effect of Cyclic Flux on Creep Rate (Low Fluence)	Significant
3.	Flux and Temperature Effects on Low-Temperature Creep	Modest
4.	Effect of Temperature Cycling on Creep	Small
5.	Effects of Helium on Creep	Small
6.	Effects of Stress Cycling on Creep	Small
7.	Nature of Deformation During Synchronized Temperature, Flux and Stress Cycles (Katcheting)	None

2. Recommendations: Update

a. Irradiation Creep and Swelling:

(1) Helium Effects

The mechanisms for irradiation induced creep and swelling deformations are better understood now as compared to several years ago. In a fusion environment, the two key factors that may modify our understanding are helium generation and neutron flux cycling. It was recommended that helium effects on irradiation creep be fully assessed. This is particularly important at low temperatures where the interaction of helium with Frenkel defects may result in an altered dislocation microstructure. Since the ratio of swelling to creep rates is mainly responsible for the presence of residual stresses, low temperature creep measurements where swelling is low are significant. A mechanistic model should be developed and correlated to low temperature measurements.

(2) Effects of Pulsing

Radiation pulsing has been recognized to change the kinetics of point defects. This was shown by various investigators to result in significant alterations in phenomena that are non-linear in their dependence on the prevailing point defect concentrations. One complicating experimental feature in pulsing experiments using ions is the inevitable temperature rise associated with the rapid electronic energy deposition during the rise time of the pulse. Experimental and theoretical efforts over the last few years have defined the conditions under which pulsing effects are expected to be pronounced. It is recommended that further experiments and calculations are performed for the following conditions:

1. Good temperature control during a rapid pulse rise time
2. Low temperature
3. Long pulse on-time, and different duty factors
4. Microstructures characterized by various obstacle sizes going from very small to large.

(3) Cascade Effects

Cascade-induced irradiation creep has been introduced as a possible deformation mechanism (1,2). While experimental verifications of this proposed mechanism are difficult, experiments can be designed for various PKA energies. The sensitivity of irradiation creep to cascade size and frequency can then be studied. Theoretical and experimental efforts are recommended for the exploration of cascade effects on microstructure, and hence properties. The prevailing perception is that irradiation creep is a prime target for such efforts.

b. Creep-Rupture

Creep-rupture is a clearly defined failure mechanism for structural components operating at high temperatures which may be further enhanced by a steady displacement damage rate. The importance of this failure mechanism is increasing, since fusion reactor concepts are emphasizing steady-state rather than pulsed operation. During the previous meeting (3), fatigue crack growth was emphasized. This emphasis is now shifting to creep crack growth and creep rupture processes.* Although thermal creep phenomena have been at the core of mechanical metallurgy studies for decades, the influence of irradiation on these processes is not fully established. Curious experimental findings have been recently reported on the effects of irradiation on the rupture time. The results seem to be at odds and the entire problem needs to be assessed for fusion application. The following evaluations are recommended:

1. The differences between post-irradiation and in-reactor creep-rupture behavior.
2. The effects of helium generation and migration on intergranular cavity nucleation and growth and fracture processes.

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*This is one is also discussed under time-dependent fracture p. 25; however, it was discussed in this section with respect to the effect of flow processes on creep-rupture.

3. The influence of irradiation on the behavior of solid solution and second phase hardeners is required, since creep strength is derived from the presence of key elements both in solution and in the form of dispersed phases.

4. The effects of impurity segregation on creep ductility.

5. The effects of flux cycling and kinetic phenomena on creep-rupture.

Although creep-rupture has been modelled in the last few years, the particular aforementioned aspects pertaining to fusion were not thoroughly addressed.

C. Irradiation Hardening and Embrittlement

It was generally shown that a significant improvement of our understanding of hardening mechanisms has resulted over the last few years. Such an understanding should be strengthened even further. However, important as it is, irradiation hardening does not immediately translate to design failure criteria. Relations are known to exist between hardening and embrittlement. This is particularly true for ferritic alloys where the yield and fracture stresses can be related to the shift in the DBTT. The following were the group recommendations:

1. Continue to establish correlations of radiation-hardening with microstructure. In this regard the following aspects were felt to be important.

   (i) Effects of the impurity content on the mobility of self-interstitials at low temperature and hence on clustering.

   (ii) Design experiments based on the theory of radiation hardening to investigate the roles of short-range versus long-range dislocation interaction with obstacles.

   (iii) Assess the importance of the number of lobes created within a cascade region on measured hardening.

2. Continue assessment of the role of helium in hardening and/or softening of irradiated metals.

3. Correlate hardening measurements with measurements of the DBTT.

4. Investigate the implications of hardening saturation and softening on the DBTT of ferritics.

5. Assess the effects of flux on the DBTT.

6. Continue developing tensile-indentation correlations.

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B. EFFECT OF FLOW PROCESSES ON FRACTURE Discussion Leader, W. G. Wolfer, Univ. of Wisconsin

1. Introduction

The connection between plastic flow properties and fracture behavior is of particular importance to the alloy development effort for fusion applications for the following reasons:

a. Experiments in FMIT will be limited by a relatively small irradiation volume, and therefore, small specimens are desirable.

b. Fracture parameters can be measured only on relatively large samples, whereas valid tensile data can be obtained from microtensile specimens.

c. Lifetime analysis of first-wall components has demonstrated that radiation-induced embrittlement causes the fracture toughness to drop and thereby reduce significantly the number of fatigue cycles to failure. The recognition of the importance of fracture behavior on the lifetime was the impetus in the previous workshop to consider the feasibility to predict fracture properties from other tensile properties. Accordingly, the following recommendations were made two years ago:

(1) Perform Fracture Toughness Experiments on Specimens of Highly Irradiated Duct Material (high priority).

(2) Perform Fatigue Crack Growth Experiments on Specimens of Highly Irradiated Duct Material (high priority).

(3) Initiate Theoretical and Experimental Studies to Identify the Microstructural Causes and Conditions for Channel Deformation (high priority).

(4) Evaluate Ductile and Brittle Fracture Models (medium priority).

(5) Develop a Standard Lifetime Analysis Code for the First Wall (low priority).

In the following, we will discuss what action has been taken and what progress has been made regarding the above recommendations.

2. Status Report on Previous Recommendations

a. Perform Fracture Toughness Experiments

Although no direct action was taken by DAFS supported investigators, some experimental results have been obtained by Huang and Fish (1). These investigators performed tensile tests on notched and unnotched specimens made from 20% CW 316 duct material irradiated in EBR-II to fluences greater than 4 x 10^{22} n/cm^2 ($E > 0.1$ MeV). From the load-displacement curves from two notched specimens (with, fluences of 7.8 x 10^{22} n/cm^2) they obtained J_{IC}-values and corresponding fracture toughness values between 57.2 and 67.7 MPA $\sqrt{m}$.

In contrast, tne value of K_{IC} when estimated from the tensile data of the notched specimens according to the Hahn-Rosenfield correlation gave a value of 22.5 MPA $\sqrt{m}$. This is too low by roughly a factor of two, a discrepancy also found by Wolfer and Jones (2) in their prediction of fracture toughness from tensile properties of HFIK-irradiated 316 samples. Their analysis showed that a correlation due to Krafft (3) and modified by Schwalbe (4) gave more reasonable predictions.

It is therefore of interest to apply the Krafft relationship

$$K_{IC} = \frac{\sigma_Y}{(1 - 2\nu)} \{\pi(1 + n)d^*(\varepsilon_f E/\sigma_Y)^{1+n}\}^{1/2} \tag{1}$$

to the experimental data of Huang and Fish. In their experiment, the measured yield strength was σ_Y = 530.8 MPa, the strain hardening exponent was n = 0.035 and the true strain at failure was ε_f = 0.3 as obtained from an estimate of the reduction in area.

If the dimple spacing d^* is assumed to be between 30 μm and 50 μm, the range of the grain size, then the computed values for K_{IC} would be between 136 and 176 MPa $\sqrt{m}$ Clearly, these values are too large.

Although Schwalbe (4) suggests originally that ε_f should be the true fracture strain, it is more in accordance with the spirit of the Krafft model to take for ε_f the tensile elongation of smooth tensile specimens as recommended by Wolfer and Jones *(2)*. Based on the measured values on irradiated 20% CW 316 duct material, the tensile elongation at test temperatures above 550°C is about 0.05 (1). Using this value for ε_f, one predicts a fracture toughness between 53.8 MPA $\sqrt{m}$ and 69.4 MPa $\sqrt{m}$ This compares very favorably with the measured range which was between 57.2 and 67.7 MPa $\sqrt{m}$.

In conclusion, the experimental results for the fracture toughness on irradiated duct material have so far demonstrated two important points relevant to the fusion material development efforts:

1) Fracture toughness of austenitic steels can indeed remain very adequate even after high fluence irradiations.

2) It appears that a satisfactory correlation has been found to predict K_{IC} from tensile data of irradiated specimens, provided the mode of fracture is transgranular and at least partly ductile (i.e., with a dimpled fracture surface). In spite of these encouraging results, however, there remain unanswered issues related to channel fracture which will be discussed below.

b. Perform Fatigue Crack Growth Experiments on Specimens of Highly Irradiated Duct Material

No action has been taken on this recommendation, and no experimental investigations have been performed in other programs.

However, a thorough review of the literature on fatigue crack growth (FCG) has been performed by Lucas and Ritchie (5), Watson (6), and Wolfer and Jones (2).

Lucas and Ritchie (5) reviewed the threshold effects on FCG. The threshold stress intensity for FCG at an R-ratio equal to zero is known to decrease with increasing yield strength. Figure 1 shows this empirical correlation for high strength martensitic steels (triangles). For austenitic steels, insufficient data prevent a similar conclusion to be drawn at the present time. The effect of the increasing R-ratio on the threshold stress intensity factor ΔK_o is shown in Figure 2.

In inert environments such a vacuum, helium, or pure liquid sodium, higher threshold values are measured than for air. Low temperature steam (260°C) was also found to increase the threshold in 430 SS and even decrease the FCG rate in Stage 11.

Very short cracks of depth < 0.5 mm are known to grow faster than predicted with empirical correlations based on FCG of larger cracks. A short crack correction can and should be made when results are applied to thin walls. Based on the recent review by Hudak (7), Watson (6) recommended that the actual crack length, a, be replaced by $(a + \ell_o)$, where

$$\ell_o = \frac{1}{\pi}\left[\frac{\Delta K_o}{\Delta\sigma_e}\right]^2$$

and $\Delta\sigma_e$ is the endurance limit for smooth, uncracked fatigue specimens. For 316 SS, a crack is effectively enlarged by $\ell_o = 0.064$ mm.

Based on the present understanding of the factors which influence the threshold for FCG, radiation damage is expected to harden structural materials and thereby reduce the threshold ΔK_o. However, other environmental effects may dominate the threshold value of ΔK_o.

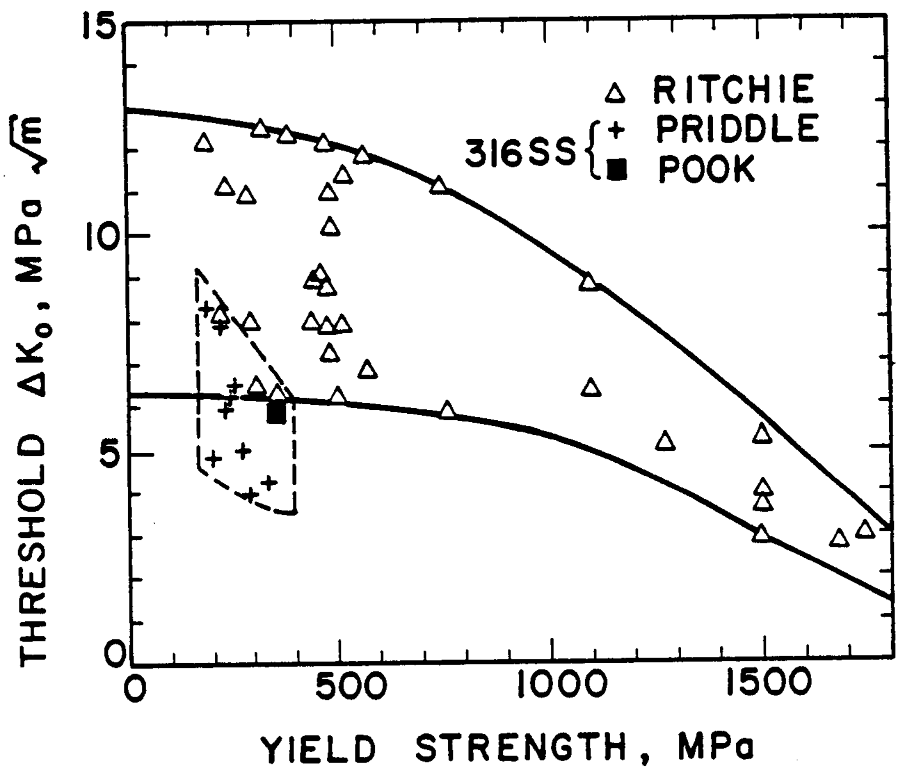

FIGURE 1. Trend for the Variation of the Threshold ΔK_o for Fatigue Crack Growth in Steels at R = 0.

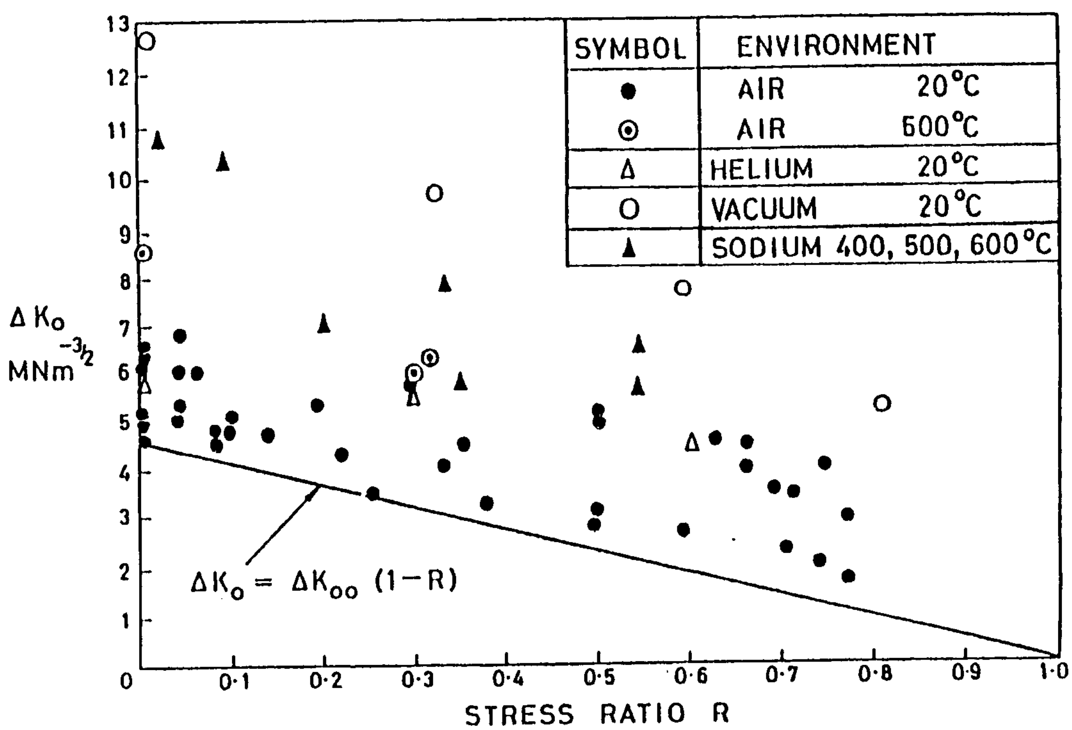

FIGURE 2. Effect of R-Ratio on Threshold Stress Intensity Factor for 316 SS.

c. Initiate Theoretical and Experimental Studies to Identify the Microstructural Causes and Conditions for Channel Deformation

Channel deformation is a particular form of flow localization which occurs in irradiated materials. It is associated with a yield strength approaching the ultimate strength and with a very small tensile elongation. It is also believed to cause transgranular failure in a non-ductile manner resulting in a cleavage-like fracture surface without the dimples characteristic of ductile failure.

Channel deformation and channel fracture presents two primary concerns to the fusion materials development effort:

1) The usual correlations between fracture parameters and other tensile properties are not applicable to channel fracture. No other established correlations or fundamental understanding existed previously regarding this fracture mode.

2) Microstructural conditions are likely to be obtained in structural materials exposed to the fusion environment which favor both channel deformation and channel fracture. It is therefore important to find out what failure design criteria are appropriate for this kind of fracture mode.

Some progress has been made with regard to these issues both in the experimental and the theoretical area. Unpublished results by J. J. Laidler (8) on the microstructure in channel deformation bands of EBR-II ducts made of 304L stainless steels have been analyzed by F. A. Garner. The excessive plastic shear in the channels not only led to sheared voids but also to a drastic reduction in the loop and the dislocation network density. It appears then that the flow localization is due to a plastic instability caused by work-softening. The work-softening is due to the coalescence of glide dislocations with their main glide obstacles, namely dislocation loops, and not due to the shearing of voids.

This interpretation agrees also with the negative results of a theoretical investigation by Wolfer and co-workers *(9)*. In these investigations, the strain energy plus the surface energy of sheared voids was evaluated as a function of the applied stress and as a function of the shear deformation in the channel. With increasing shear, the surface energy of the sheared voids increases, but the strain energy decreases. An instability point for a critical value of the shear was anticipated to occur, and it was thought that this might explain the onset of channel deformation. If correct, a close connection between the amount of void swelling and the onset of channel fracture would have existed. However, the numerical results showed that an instability of this kind does not occur for void sizes of any practical interest. Therefore, there does not seem to exist a direct link between void swelling and channel deformation, and the flow localization may have a different microstructural origin. However, it is possible that voids contribute indirectly to the channel fracture phenomenon. The observation by Laidler seem to implicate dislocation loops as the important microstructural feature. A detailed correlation between the loop density, size, etc. and the onset of channel deformation is, however, still lacking.

Some progress has also been made in the development of models which relate microstructural parameters for channel deformation to fracture toughness. Wolfer and Jones (2) further developed a model proposed earlier by Smith, Cook, and Rau (10). This model relates the fracture toughness to the channel width, the yield stress, and to the shear decohesion displacement in a channel. Among these parameters, only the yield stress and perhaps the channel width can be obtained at the present time from small tensile samples or microspecimens. The shear decohesion displacement could, however, in principle be measured with a die-punch experiment. This possibility needs further investigation.

When this new model for fracture toughness was applied to channels observed in EBR-II duct material of 304 L, a lower bound of about 30 MPa$\sqrt{m}$ was found. The actual fracture toughness for this rather brittle material must then be higher. This implies that the fracture toughness of this severely radiation-embrittled material is still adequate in spite of its very low ductility.

To summarize, the nature of channel deformation has been clarified to some extent, and a model for the fracture toughness has been developed for channel fracture. This model suggests again the possibility of obtaining fracture parameters from microtensile specimens. However, further development is required to realize this potential.

d. Evaluate Ductile and Brittle Fracture Models

Progress regarding this recommendation is discussed in connection with recommendations (a) and (c). It should be pointed out, however, that none of the fracture models evaluated and developed apply to ferritic and martensitic steels and to the issue of the DBTT. With the rising interest in ferritic and martensitic steels, more development is required.

e. Develop a Structural Lifetime Analysis Code for the First Wall

Significant progress has been made in this area by the University of Wisconsin group. A set of codes has been developed which follow the stress history of first wall and in-vessel components exposed to the plasma heat and particle flux as well as to the neutron flux. Changes in the material's properties are modeled as a function of exposure time, and the initiation and propagation of fatigue cracks is followed. As a result, self-consistent lifetime predictions can be made.

Because of the lack of essential radiation damage data, however, the present predictions are somewhat hypothetical. However, the codes developed allow researchers in the alloy development effort to identify important and relevant areas of research, and to test the impact of correlation for mechanical property changes.

Extensive lifetime predictions have been made by Watson (6) on a blanket module in a tokamak reactor; the structural material used was 20% CW 316 stainless steels. Similar work is in progress for martensitic steel as a first wall material in both tokamak and tandem mirror reactors.

3. Recommendations

- Since only one measurement of fracture toughness has been reported so far, more J_{IC}-measurements on type 316 SS irradiated to high doses are recommended.
- Establish by SEM and TEM what mode of fracture occurs in the J_{IC} experiments on type 316 SS irradiated to high doses.
- Explore what further experiments could and should be done to find out the conditions when a transition occurs from transgranular to intergranular fracture. A transition of this kind was found in experiments of Huang and Fish at test temperatures of ~650°C. This transition temperature is expected to depend on dose and on the helium content.
- Collect and analyze the inforination on the transition temperature from trans- to intergranular fracture in irradiated stainless steels.
- More emphasis should be placed on the effects of irradiation on welds in 20% CW 316 and on the fracture properties of the heat-affected zone.
- Address the following questions with regard to the threshold ΔK_o in FCG:
  a) What is the effect of the environment (hydrogen, oxygen, water ...) on ΔK_o?
  b) What *is* the impact of a ΔK_o variation on lifetime?
- Analyze the impact of the softening observed in 20% CW 316 at high doses.

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C. TIME-DEPENDENT FRACTURE, Discussion Leader: G. Lucas, UCSB

1. Introduction

The status of work performed in the area of time-dependent fracture was discussed in four different categories; namely, fatigue, creep-rupture, creep-fatigue, and environmentally-assisted fracture. From discussions with members present at the working group, recommendations in each of these categories were updated. Both the status of original recommendations and the update versions are described for each category below.

2. Status and Update of Recommendations

a. Fatigue

The original recommendation in the area of fatigue was that a report be made which addresses the issue of near threshold crack growth as it pertains to fusion device design. While it was stated in this recommendation that such a report should address differences between short crack behavior (and thus crack initiation phenomena) and long crack behavior, this was emphasized in comments by Argon, Chen and Harling (MIT).

Discussion of this recommendation before the general group established the following:

- Such a report is nearly drafted. It incorporates recent reviews of fatigue crack propagation in austenitic stainless steels by James,[1] near-threshold phenomena by Ritchie,[2] and short crack behavior by Ritchie and Suresh.[3] The report will appear as a DAFS quarterly contribution and will also be submitted for publication in a refereed journal.
- The most recent data reported on fatigue behavior of candidate alloys while useful in their own right and necessary for a complete description of fatigue behavior, still leave some questions about near-threshold and short-crack phenomena. Data obtained on a variety of materials from center-cracked tension (CCT) specimens in both the unirradiated[4] and irradiated[5] conditions are for crack propagation

rates greater than 10^{-6} mm/cycle, above the threshold regime for long cracks at the reported test conditions. Also, the distinction between fatigue crack initiation and fatigue crack propagation is not apparent in the S-N data on HFIR-irradiated austenitic stainless steel.[6]

- Fatigue crack propagation in the near-threshold regime, fatigue crack initiation, and differences in the behavior of short cracks and long cracks cannot be discounted in failure analysis for fusion structures. Recent work by Watson et al.[7] points to the importance of including both near-threshold and near-K_{IC} behavior of fatigue cracks in failure analysis. Moreover, continuing work on plasma-surface interactions indicates that fatigue crack initiation may occur by non-classical mechanisms, at least on the plasma side of the first wall, thereby "short circuiting" that part of total fatigue life. Fatigue crack propagation of short cracks may then become quite important.

- Environmental effects on fatigue crack propagation, especially in the early stages of crack propagation may be of primary importance. This is certainly the case in long-crack, near-threshold behavior as well as recent short-crack studies, although considerably less is known about environmental effects on short crack behavior.

Consequently, the recommendation remains that the topical report/recommended research program on fatigue considerations for fusion device design be completed and circulated for review and comment prior to distribution.

b. Creep-Rupture

Of the recommendations in the area of creep-rupture the first was that calibrated correlation methodology (CCM) be continued. The CCM approach has received continued attention at UCSB. Odette and Vagarali[8] have reported results of modeling efforts to incorporate the influence of He bubbles on post-irradiation creep rupture times in austenitic stainless steels. The model is based on an assumed exponential size distribution of pre-existing

He bubbles at grain boundaries, which is calibrated against microstructural evidence, and bubble growth to a critical areal density by coupled grain boundary diffusion and matrix creep, after Edward and Ashby.[9] The model predictions are in good agreement with existing data.

While work has continued on CCM development, further development is clearly needed. The physical basis of the modeling of Odette and Vagarali represents but one of several possible competing mechanisms for creep rupture. Moreover, there is a need to incorporate other pertinent operating variables in the methodology; in particular, the effect of in-reactor creep rates on creep rupture life. However, to assess the effects of in-reactor creep, the data base appearing in the literature needs to be extended. F. Garner (HEDL) agreed to summarize the available Breeder data for this purpose.

Hence, the first of the recommendations on creep rupture stands modified as follows: Work should continue on the development of physically-based, calibrated correlation methodologies for creep rupture in candidate alloys; and as a first step in treating the effects of in-reactor creep on creep rupture the Breeder data on creep rupture should be summarized in a topical report.

The second of the original recommendations was that pressurized-tube specimens should be flagged in a systematic manner for microstructural analysis to provide a consistent data base for CCM development. Although a specific DAFS plan has not been implemented within the ADIP test plan, a large number of the pressurized-tube samples being tested are also targeted for microstructural analysis. Consequently, the original recommendation is maintained with the understanding that it is already being largely implemented by ADIP. There may be a need to augment the ADIP specimens with DAFS specimens in the future; however, such a need will be dictated by CCM developments.

The third of the original recommendations was that alternate creep/creep rupture specimen designs should be investigated for the purpose of studying stress state effects on creep rupture. Schwab[10] has investigated an eccentric tube design for such purposes and has found some success. However, there is still a need for other specimen designs, and therefore the recommendation stands.

Finally, the fourth of the original recommendation was that: the literature on creep crack growth should be monitored and pertinent developments should be incorporated in creep rupture analysis. This is being done by groups involved in creep rupture analysis, but as yet creep crack phenomena have not been incorporated in CCM's. Therefore this recommendation is maintained.

With respect to new recommendations, there was some discussion about embrittlement of 316 stainless steel irradiated and tested at an elevated temperature. Huang[11] reported results on plane strain fracture toughness (K_{IC}) of 316 stainless steel irradiated in EBR-II at 377-400°C to 11.3 x 10^{22} n/cm^2 E_n > .1 MeV. While the fracture toughness was –70 MPa$\sqrt{m}$ and the fracture mode predominantly transgranular for test temperature <450°C, the fracture toughness dropped to 21 MPa$\sqrt{m}$ and the fracture mode was completely intergranular at 650°C. Such a loss *of* fracture toughness is considerable and warrants further attention. The mechanism for the high temperature fracture is not currently known; possibilities include He embrittlement as well as impurity segregation. Although the temperatures at which this phenomenon was exhibited were those in which irradiation damage and He effects on creep rupture have been largely investigated, the strain rates involved in the K_{IC} tests were considerably higher. Hence, there may or may not be a relationship between this loss of fracture toughness at high temperatures in irradiated 316 stainless steel and reduction of creep rupture life,

Nonetheless, this exhibition of elevated temperature fracture toughness loss warrants further investigation; and from a time-dependent fracture point of view the recommendation for research in this area can be stated *as* follows:

The effect of strain rate on irradiation embrittlement should be investigated with emphasis on the relation (if any) between loss of fracture toughness in irradiated austenitic stainless steels at elevated temperatures.

c. Creep-Fatigue Interaction

The original recommendation in the area of creep fatigue was that critical tests should be performed to evaluate the potential for He to alter the creep-fatigue interaction, and additional research, if needed, should thereafter be suggested.

Such critical tests have yet to be performed. Moreover, creep-fatigue interaction is still a potential problem, although of lower priority than the others identified here. Consequently, the original recommendation is maintained.

d. Environmentally-Assisted Fracture

The original recommendation in the area of environmentally assisted fracture was that DAFS should designate a liason with other components of the fusion reactor materials program to ensure that chemically-induced fracture considerations be made when and where necessary and to ensure that the assistance of DAFS is provided when necesssary in the accumulation of data and the understanding of problems of this nature.

R. Jones (PNL) agreed to serve as the liaison and has kept both DAFS and ADIP aware of other's activities in this area. Perhaps it is unrelated, but it is interesting to note that in the recent ADIP exchange meeting it was determined that environmentally-assisted fracture should be a priority consideration for alloy development, particulary for stainless steels in water environments. Consequently, it was generally agreed that the liaison be continued.

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D. RADIATION EMBRITTLEMENT AND ENVIRONMENTAL EFFECTS. Discussion Leader, R. H. Jones, Battelle Northwest Laboratory

1. Introduction

The scope of the embrittlement activity was changed substantially from the previous workshop. The name was changed to reflect the greater emphasis on stress corrosion and corrosion fatigue processes while the topic of low temperature (<300°C) helium embrittlement of austenitic stainless steel was switched to the time dependent fracture section. While helium embrittlement is certainly a form of radiation embrittlement, the switch was made to reduce duplication since helium embrittlement was covered under the time dependent fracture section.

Included in the radiation embrittlement and environmental effects section is radiation enhanced impurity segregation, and precipitation which can contribute to embrittlement, hydrogen embrittlement, stress corrosion and corrosion fatigue. Property-property correlations are also included even though some of this activity is directed towards flow and time dependent fracture properties.

Property-property correlations generally refer to attempts to use one set of material properties for predicting another set of properties which are not available or attainable because of experimental limitations. Examples include the use of micro-hardness measurements to predict flow properties or the use of tensile properties to predict fracture toughness. Property-property correlations may be based on experimental data or may be a blend of experimental data and modelling. The need for property-property correlations results from the lack of adequate fusion energy neutron irradiation facilities for irradiating samples from which the desired properties can be obtained.

2. Recommendations: Status and Update

a. Austenitic Stainless Steels

(1) Evaluate the Role of Helium on Ductility at Low Temperatures

It was concluded at the workshop that the effect of helium on ductility at low temperatures should be included with the section on the effects of helium on time dependent fracture since helium effects are a primary factor in the time dependent fracture of materials and because the transition from time dependent to rapid or time independent fracture is not well defined.

The questions raised in the previous workshop report (DOE/ER-0046/4 Vol. 2, p. 44) have not been answered during the last two years. While these questions, listed below,

- is atomic helium at grain boundaries contributing to the embrittlement;
- if so, what is the relationship between quantity of helium at the grain boundary, yield strength and embrittlement;
- also, how is helium transported to the grain boundaries;
- is helium affecting flow properties, i.e., flow localization, and, therefore, ductility?

are of primary concern to dustenitic stainless steels they could also have application to other alloy systems. Therefore obtaining answers to these questions is still considered to be high priority.

(2) Develop Property-Property Correlations (High Priority)

(a) Experimental Correlations

Comparison between microhardness and yield strength of copper[1], titanium and vanadium[2] irradiated with T(d,n) neutrons has been completed in the last few years. The results shown in Figures 1 and 2 indicate that a good correlation exists between the microhardness and tensile yield strength of copper irradiated to low fluences at room temperature but that a simple correlation does not exist for titanium and vanadium. The breakdown in the correlation for titanium and vanadium may be the result of non-uniform flow around the microhardness indentor. These results suggest that property-property correlations should be established for each alloy system and over the irradiation fluence and temperature range of interest before an extensive amount of microhardness data is acquired. A microhardness-yield strength correlation is needed on irradiated austenitic stainless steel and HT-9 before microhardness data can be used with confidence for these materials.

An indentation creep and load relaxation test was developed[3] and the results were found to compare favorably with standard creep and stress relaxation results for lead. Comparison between the indentation and standard method was made over strain rates covering about 3 orders of magnitude. Applicability of this techinque to austenitic stainless steels needs to be assessed; however, the significance of post-irradiation creep rate for fusion reactor design is not clear and therefore the value of this technique is uncertain.

(b) Model Based Correlations

Both fracture toughness and fatigue crack growth rate model predictions have been evaluated recently. The calculated change in fracture toughness, K_{IC}, with irradiation fluence is shown in Figure 3 based on the Krafft model and tensile data from HFIR[4] and EBR-II[5] irradiated austenitic stainless steel. The results given in Figure 3 are very comparable with the high He/dpa HFIR environment giving slightly lower toughness than the low He/dpa EBRII environment. Further development of a model-based fracture toughness correlation must await experimental data to assess the quality of'these calculations. The value of such a correlation is in the possibility of showing the trend in fracture toughness from simple mechanical property data such as tensile test data which is more readily obtainable with existing fusion energy neutron devices.

A model based fatigue crack growth rate correlation has also been evaluated. The McEvily[6] relationship was used[5] in conjunction with the Krafft K_{IC} model to predict the da/dn vs $\phi\tau$ relationship for austenitic stainless steel. The results given in Figure 4 for R values of 0 and 0.3 suggest that the fatigue crack growth rate of 316 SS initially increases with increasing fluence but that the fatigue crack growth rate is independent of fluence for fluences greater than 3 $MW\text{-}yr/m^2$. As for the fracture toughness correlation, further development of the fatigue crack growth rate model must await experimental verification. It is recommended that further development of model-based correlations be continued with a major emphasis of comparing predicted and experimental results whenever possible.

(3) Develop Techniques for Measuring Grain Boundary Helium and Impurities

(a) Grain Boundary Helium (Medium Priority)

There have been no published reports or papers on this topic and I am not aware of anyone attempting to develop a technique for measuring the helium concentration in the grain boundaries of irradiated material. Residual gas analysis, HGA, of the quantity of helium released upon fracture is the easiest approach. An RGA/Fracture capability exists at Oak Ridge National Laboratory; however, very few results have been published. The helium release upon fracture technique has the limitation in irradiated materials containing voids or bubbles at the grain boundary that the major helium release is from the voids or bubbles. Ihus the quantity of helium adsorbed at grain boundaries is not detected. However, the RGA/Fracture technique would be very useful for materials which do not contain grain boundary voids or bubbles and would be a useful starting point for those that do contain grain boundary voids or bubbles.

(b) Grain Boundary Impurities (High Priority)

There have been no published results on grain boundary chemistry of irradiated material within the last 2 years. Some work done several years ago[7] showed that irradiation increased the grain boundary phosphorus concentration of 304 SS; however, the irradiated samples were fractured at elevated temperatures. Desegregation or segregation could occur during the elevated temperature fracture process thereby causing great uncertainty in the reported phosphorus concentrations.

Surface segregation studies have been used extensively to evaluate radiation induced segregation of minor alloying elements; however, this technique has not been used for studying the effects of radiation on impurity segregation. Significant differences between surface and interface segregation behavior for thermally activated-equilibrium segregation and the need to expose the surface to the atmosphere between irradiation and analysis limit the usefulness of this approach; however, the simplicity of free surface composition analysis relative to grain boundary composition analysis for irradiated materials makes the free surface approach useful for scoping studies.

Scoping studies to determine the effect of irradiation on impurity segregation of 316 SS, PE16, HT-9 and Ti-6Aℓ-4V[8] have been completed this year. Significant irradiation enhanced phosphorus segregation was observed in 316 SS and PE16 while no irradiation enhanced impurity segregation was observed in HT-9 and Ti-6Aℓ-4V. The quantity of phosphorus segregation on the free surface of heavy ion irradiated 316 SS is shown in Figure 5. Thermal control samples had negligible free surface phosphorus segregation. If similar quantities of phosphorus segregate to the grain boundaries of 316 SS, the stress-rupture, stress-corrosion and corrosion fatigue properties could be adversely affected.

Future recommended work on this topic includes determining if the free surface phosphorus concentration saturates at higher fluences and performing AES grain boundary composition analysis of irradiated 316 SS. The higher fluence tests will reveal the concentration at which saturation occurs. This information can be used to determine if irradiation induced phosphorus segregation is non-equilibrium or enhanced equilibrium segregation.

(4) Write a Summary Report on Hydrogen Embrittlement of Austenitic Stainless Steels.

Several good summaries(9-11) have been published on this topic in the last several years and therefore it was concluded that a separate summary for the fusion program is not necessary. Hydrogen embrittlement of austenitic stainless steels in fusion reactors is very unlikely; however, a minor concern still exists. These issues will be addressed in the recommendation on environmental effects of fusion reactor materials.

(5) Evaluate Environmental Effects (High Priority)

This recommendation represents a new activity for the DAFS task group. The recommendation made at the previous workshop was for the DAFS task group to provide liaison on stress corrosion with other task groups (p.38). It has since been concluded that there is a need to identify critical environmental effects issues and to assess and develop models for stress corrosion and corrosion fatigue of fusion reactor materials. The determination of the stress corrosion and corrosion fatigue properties of fusion reactor materials is recognized as the responsibility of the ADIP Task Group while the goal of the DAFS Task Group in this activity is to identify critical effects and the development and assessment of stress corrosion and corrosion fatigue fracture models. The specific recommendations for this activity include:

1) To measure the effect of grain boundary segregation on the stress corrosion of 316 SS.
2) Assess the status of stress corrosion and corrosion fatigue models for fusion applications. Publish this status report in the DAFS Quarterly, August 1984.

b. Ferritic Steels

(1) Evaluate the Role of Trace Impurities on Embrittlement (High Priority)

Trace impurity segregation has been well established as a major factor in the embrittlement of ferritic steels. In the past two years, the following information has been acquired regarding impurity segregation in HT-9:

- sulfur may segregate to martensite lath boundaries and contribute to temper embrittlement[12]
- phorphorus segregation (0.1 monolayers) was observed at grain boundaries of HT-9 annealed for 240 hrs at 540°C[13]
- radiation enhanced impurity segregation was not observed in HT-9[8] or a Fe + 300 appm P alloy.

These results indicate the primary impurity segregation concern for HT-9 is in the areas of establishing the proper trace impurity limits and processing and fabrication thermal cycles to control thermally activated segregation. These activities are the responsibility of the ADIP task group while the recommendations for further work by the DAFS Task Group is as follows:

- Determine if radiation enhanced impurity segregation will occur in other ferritic alloys such as 2 1/4 Cr 1 Mo steel,
- Evaluate the grain boundary chemistry of irradiated HT-9 to verify the surface segregation results.

(2) Develop Property-Property Correlations

The rationale for developing property-property correlations for ferritic steels is similar to austenitic stainless steels. This activity is divided into experimental and model-based correlations; however, no progress was made in the past two years. Recommended research activities include:

- Develop micro-hardness-yield strength correlations for iron and ferritic alloys,
- Evaluate model-based correlations for fracture and fatigue crack growth.

(3) Evaluate Saturation in DBTT of Pressure Vessel Steel and its Relevance to Fusion Materials

Recent results from pressure vessel surveillance samples and some German data have indicated that the reference temperature nil ductility temperature, RTNDT, reaches a plateau at fluences of about 10^{19} n/cm^2. The light water reactor pressure vessel data are not directly applicable to fusion reactor first wall design because there are no data to show that the plateau in RTNDT persists to fluences greater than about 10^{20} n/cm^2. The LWR data may be applicable to structural components in low flux positions within the blanket of a fusion reactor; however, these must be considered on a case by case basis. Therefore, it was concluded that this is not an appropriate activity for the DAFS task group and should therefore be deleted as a recommended activity.

(4) Evaluate Environmental Effects (High Priority)

A recommendation was made at the previous workshop to "flag" the need to evaluate the hydrogen embrittlement of irradiated ferritic steels because the sensitivity of steel to hydrogen increases with increasing yield strength. This concern was confirmed for HT-9 recently be Stoltz[14] who evaluated the mechanical properties of cathodically charged HT-9 with yield strengths of 650 MPa to 1250 MPa. Stoltz varied the yield strength of HT-9 by the quenching and tempering treatment. The reduction of area was reduced from 60% to 7% for hydrogen charged HT-9 with a yield strength increase from 650 MPa to 1250 MPa. Since irradiation can double the yield strength of a material, these results confirm the need to evaluate the hydrogen embrittlement of irradiated ferritic materials.

Segregation of impurities to the grain boundaries of ferritic materials can cause a material to be more sensitive to hydrogen embrittlement. This effect was shown in Figure 6, by Kameda and McMahon[15] for a NiCr steel with Sn, or P segregated to the grain boundaries. Kameda and McMahon's results in Figure 6 show that a material which is embrittled by P and Sn segregation was further embrittled by hydrogen; however, material embrittled by Sb segregation is not further embrittled by hydrogen. Therefore, it is important to carefully characterize the grain boundary composition of both unirradiated and irradiated material and evaluate the hydrogen embrittlement if impurity segregation has occurred. Preliminary results from work at PNL[8] indicate that irradiation enhanced segregation may not occur in HT-9; however, thermally activated segregation may occur during fabrication or service.

Wright and Gerberich[16] have modeled the effect of hydrogen and impurities on the K_{TH} of the Kameda and McMahon material. The results of this analysis are in fair agreement with the experimental results as shown in Figure 7.

The recommendations for this task are similar to those for austenitic materials in that the purpose is to identify key environmental effects issues and develop models for hydrogen embrittlement, stress corrosion and corrosion fatigue of ferritic materials. Therefore the specific recommendations are:

1) Evaluate the effect of impurity segregation on the hydrogen embrittlement of HT-9,

2) If an impurity-hydrogen effect is observed, model and experimentally measure K_{TH},

3) Evaluate models for the stress corrosion and corrosion fatigue of ferritic materials.

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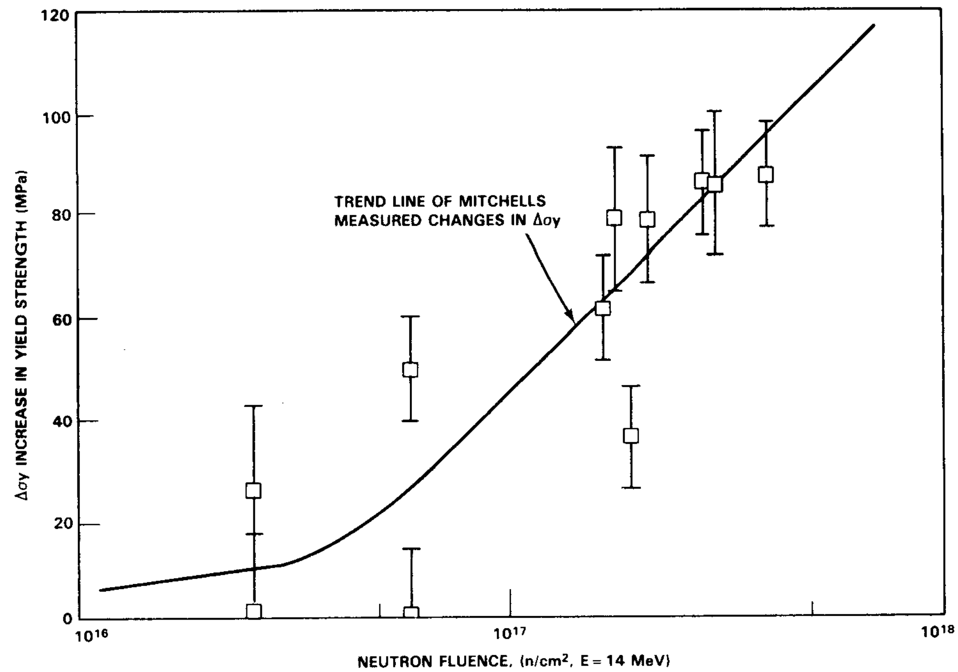
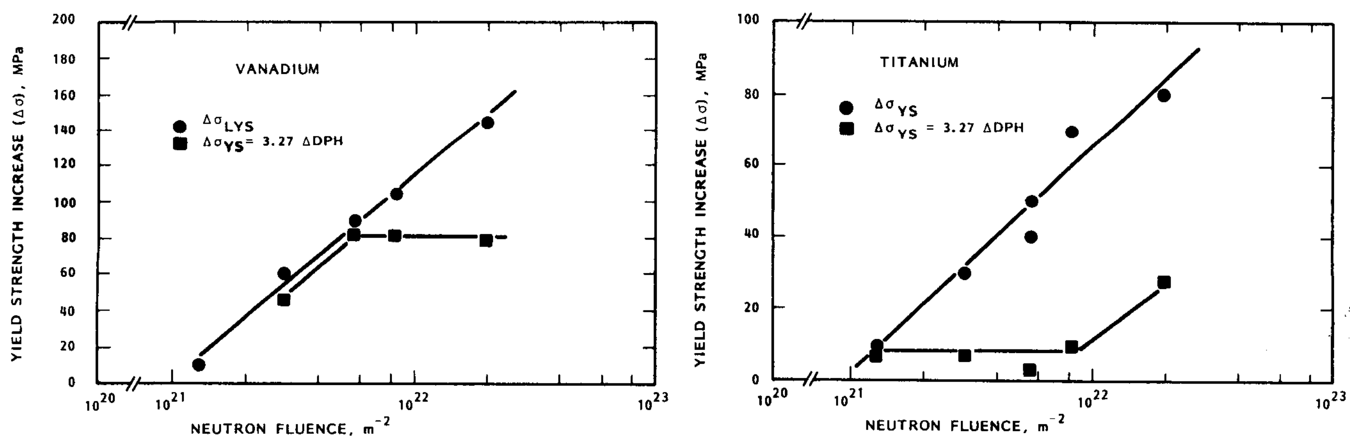


FIGURE 1. Yield Strengh - Microhardness Correlation for Copper Irradiated with T(d,n) Neutrons.(1)



(a) (b)

FIGURE 2. Yield Strength - Microhardness Comparison for T(d,n) Neutrons Irradiated: a) Vanadium and b) Titanium.

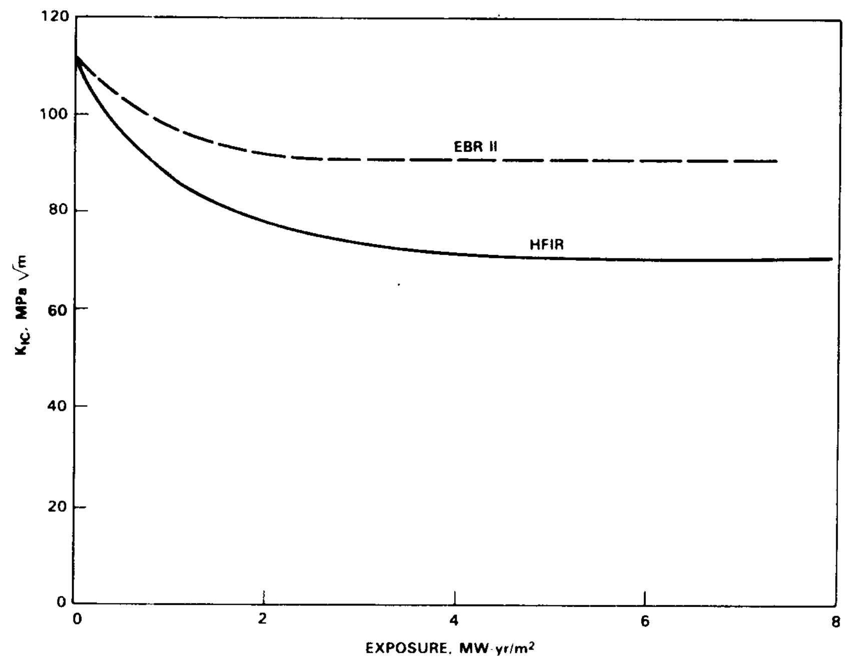


FIGURE 3. Calculated Fracture Toughness of 316 SS Using the Kraff Model and Tensile Data from EBR-II and HFIR-Irradiated Material.

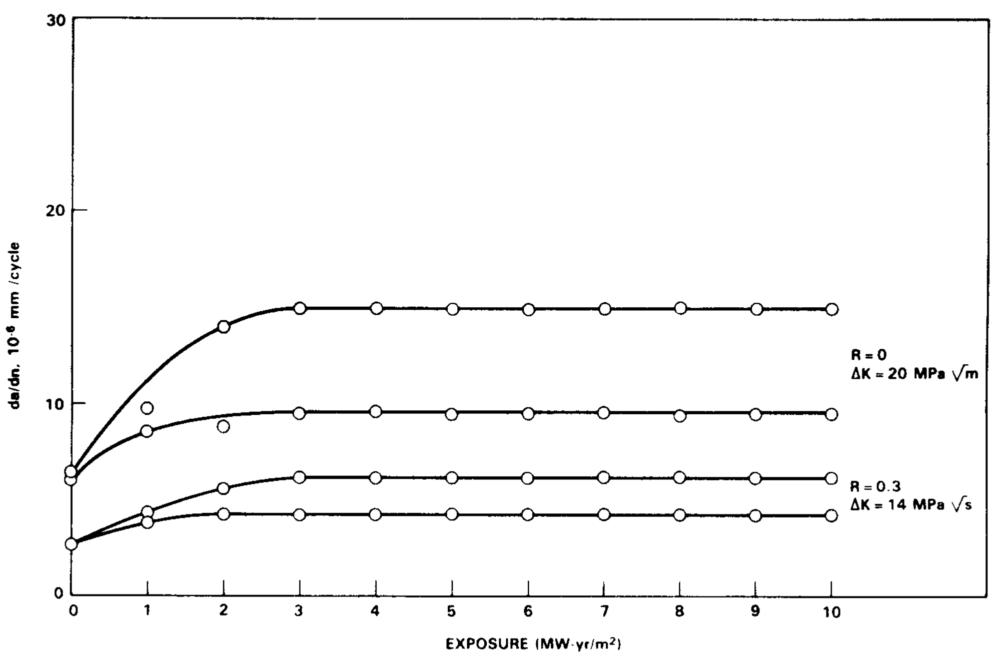


FIGURE 4. Calculated Fatigue Crack Growth Rate of 316 SS Using the McEvily Model and Tensile Data from EBR-II-Irradiated Material.

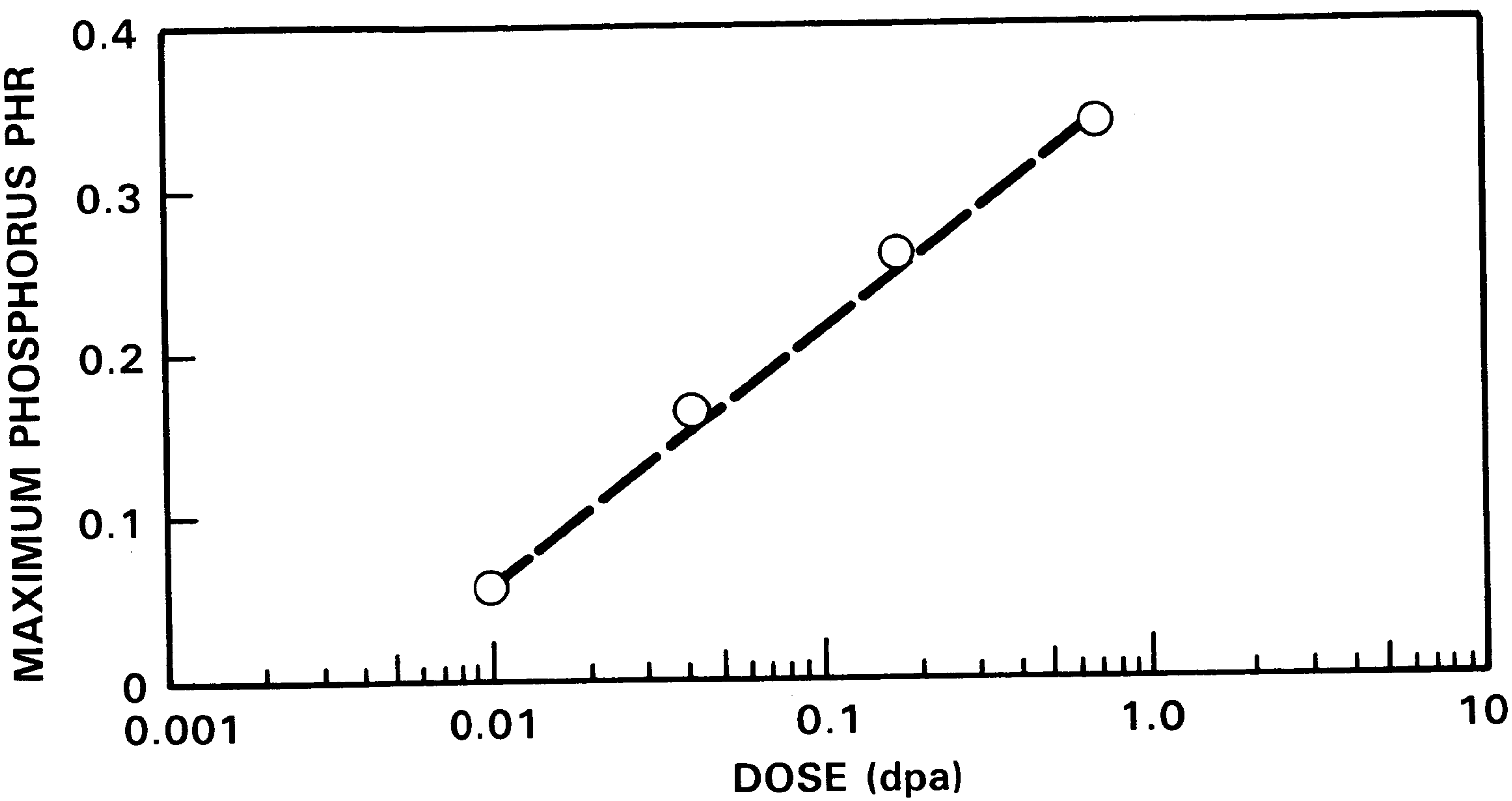


FIGURE 5. Dose Dependence of Phosphorus Segregation in 316 SS at 600°C.

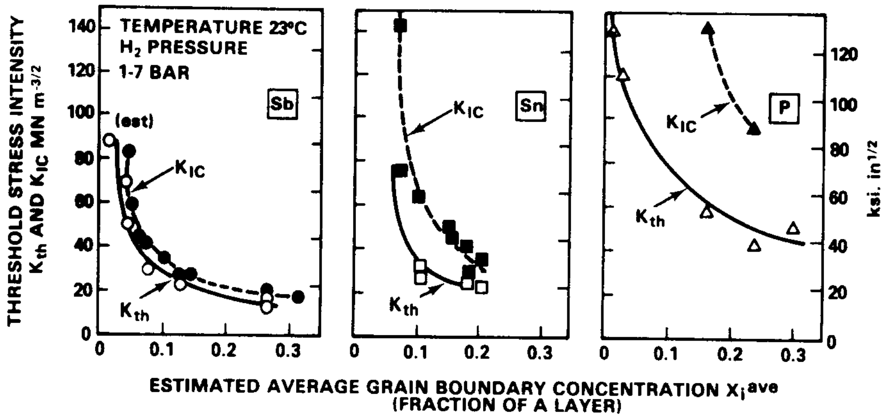


FIGURE 6. Effect of Grain Boundary Concentration of P, Sn, and Sb on K_{IC} and K_{TH} of a Ni·Cr Steel.

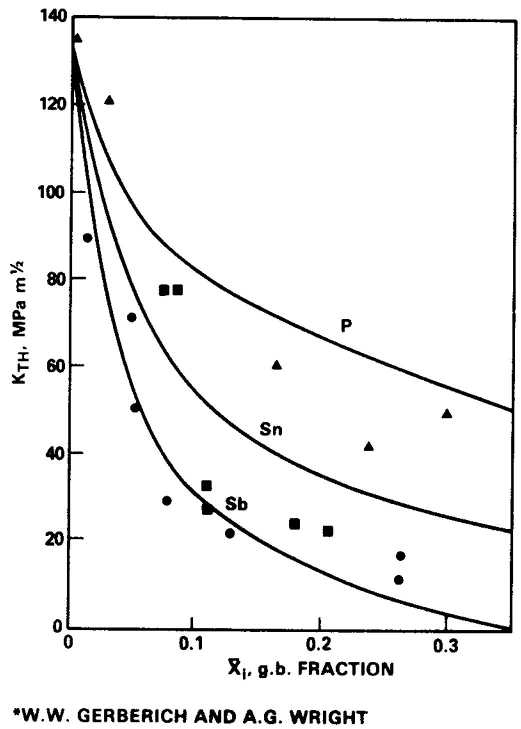


FIGURE 7. Comparison of Calculated and Measured Values of K_{TH} for a Ni·Cr Steel with Grain Boundary Concentration of P, Sn, and Sb.

APPENDIX

The information contained in the appendix was presented at the workshop but there was insufficient time for the participants to discuss the merits or details of these recommendations. Therefore, they are included in the workshop report to communicate the proposed recommendations and to encouraye discussion by the fusion reactor materials community. Written comments are encouraged and are to be sent to R. Jones.

APPENDIX

PROPOSED RECOMMENDATIONS

(Comments Requested)

A. PROPOSED EFFORT ON ADIP PATH B ALLOYS

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1. Background

The ADIP task group recently recommended that the entire Patn B alloy class be dropped from consideration as first wall and blanket structural materials. These alloys are Fe-Ni-Cr alloys strengthened by precipitation of a second coherent phase and were thought to offer promise relative to 300 series stainless steels. They were expected to have high strength, low swelling and creep, low fatigue crack growth rates and improved thermal expansion characteristics. It was also forecast, however, that the primary disadvantage of this alloy class would be their lower ductility.

Irradiation appears to accentuate this disadvantage, however. After irradiation in HFIR to 10 dpa the ductility of these alloys was assessed using disk bend testing. The ductility decreased rapidly with increasing test temperature, approaching zero at temperatures ≥550°C.[1] Some improvement in ductility was noted for some low temperature preirradiation aging treatments.

Microstructural examination of these alloys revealed that the ductility reduction was not dependent on helium level but arose from weakening of grain boundaries by formation of γ' boundary films or other larger precipitates which serve as weakening sites. These conclusions are similar to those reached by Yang[2] for Nimonic PE16 and Vaidyanathan et al.[3] for a series of alloys similar to those of Path B. In these studies however, the alloys were irradiated in EBR-II where the helium levels generated are much lower.

The ADIP task group has no plans to test this alloy class further but has suggested that additional work would be appropriate if conducted by the DAFS task group. Additional specimens are available at 20-40 dpa which can be used to confirm the ductility loss and establish the mechanisms responsible for the degradation. There is some potential for reinstating the Path B alloys if the causitive mechanism can be identified and is found to be sensitive to preirradiation thermal-mechanical history.

2. Recommendation

The DAFS task group should consider subjecting the remaining specimens to disk bend testing and microstructural examination to assess the potential of reinstating some of the Path B alloys to the ADIP effort.

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B. PKOPOSEU EFFUKT ON IKKAUIATION EFFECTS ON PKECIPITATION AND EMBRITTLEMENT OF FERRITIC MATERIALS

D. S. Gelles
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1. Backyround

In my opinion,[1,2] the major effect of irradiation on flow and fracture processes in ferritic alloys is due to precipitation. The intent of my presentation at the Workshop meetiny was to emphasize this point. I believe this point of view has not been addressed in the report on the Workshop, whereas it should have been.

As demonstrated in austenitic alloys, major consequences of irradiation are irradiation-enhanced diffusion and solute segregation. Irradition-induced precipitation may also occur. The same processes can be expected in ferritic alloys. There is now clear evidence for major microstructural changes due to precipitation in ferritic alloys followiny reactor irradiation. It is likely that precipitation is directly responsible for DBTT shifts resulting from irradiation in martensitic stainless steel[3], and it can be speculated that similar processes control DBTT shifts in pressure vessel steels.

2. Recommendation

Therefore, I recommend that consideration be given to redirect the work on flow and fracture processes in ferritic alloys in the DAFS program to include study of the behavior of irradiation-enhanced diffusion, irradiation-induced precipitation and solute segregation in ferritic alloys with regard to in-reactor precipitation. (Note: This recommendation has been included in Section D 6 p. 3.)

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C. HIGH HEAT FLUX COMPONENTS

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The Office of Fusion Energy (DOE) has recently formed a new task group called the High Heat Flux Component Task Group (HHFC). It's responsibilities are the research and development of components that receive very high heat and particle fluxes, such as neutral beam dumps, actively cooled pumped limiters, divertor collector plates, RF antenna structures, and mirror machine direct convertors. Because these components will also be subjected to neutron irradiation, the DAFS program will play an important role in the selection of materials for HHF applications.

In many respects, high heat flux components operate in a considerably more severe environment than the first wall and blanket. Surface heat fluxes typically range from 200-300 W/cm^2 for a full toroidal belt limiter[1-3] to 2,000-20,000 W/cm^2 for a neutral beam dump[4-5]. In contrast, a first wall might only see 10-50 W/cm^2, orders of magnitude less. The intense particle bombardment which causes the high heat flux can also induce severe erosion of the surface due to sputtering and other processes. This leads to conflicting design requirements because a thin wall (1-5 mm) is desirable in order to minimize thermal stresses, while thick walls (1-2 cm) are required to provide an adequate erosion lifetime.

At this time, the preferred design for high heat flux components consists of a thick, erosion resistant, possibly low-Z cladding or coating (armor) attached to a thin-walled, high thermal conductivity, actively cooled metal structure (substrate). An example might be an array of 1 cm thick SiC tiles bonded to a copper alloy plate that has internal coolant channels of 2 mm wall thickness. Two primary concerns of the HHFC task group are (1) the integrity of the bond between the armor and substrate, and (2) the lifetime of the actively cooled substrate. Although these problems for near-term reactors (TEXTOR, TFTR, JT-60, JET, and TORE-Supra) are caused mostly by thermal fatigue, it is clear that radiation damaye in future reactors will have a considerable impact on these components.

The DAFS task group needs to broaden its program to include the new materials required for high heat flux components. These "Path E" materials would include the high conductivity substrate metals: copper, molybdenum, tantalum, vanadium and aluminum, as well as armor materials: SiC, Be, TiC, TiB_2, B_4C, etc. The austenitic and ferritic steels are not considered as feasible substrate materials because their low thermal conductivity causes unacceptably high temperatures and large thermal stresses. For example, a 316 stainless steel thin-walled tube with a uniform surface heat flux of 1,000 W/cm^2 could only have a wall thickness of 0.1 mm to prevent the surface from yielding due to thermal stresses. On the other hand, a high strength dispersion strengthened copper alloy such as AMAX-MZC or AMZIKC, could have a wall tnickness of 8 mm, for the same conditions. Other alloys that have been considered are Ta-5W[2] and V-15 Cr-5 Ti[3].

Although copper alloys are the preferred choice for high heat flux substrates, very little is known about the effect of neutron irradiation on the mechanical properties, according to a review paper by Harling[6]. Knoll[7] has studied the swelling of heavy ion irradiated copper alloys, however, irradiation creep, embrittlement and fatigue crack growth are phenomena which have not been investigated. Recently, a number of samples (see Table 1) that are candidate materials combinations for high heat flux components[8] have been placed in FFTF and EBR-II. These are 1 cm thick, 1.5 cm diameter disks that will be removed near the end of 1983. The primary purpose of these tests is to study the effect of neutrons on the bond integrity. However, additional experiments need to be done using specimen geometries that are suitable for mechanical properties testing. For this reason, it is important that DAFS and the HHFC task group work together to develop a technical plan for damage analysis and mechanical behavior for these new Path E materials.

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TABLE 1

NEUTRON IRRADIATION STUDIES OF HIGH HEAT FLUX MATERIALS

COATINGS/SUBSTRATE	SAMPLES IN FFTF	SAMPLES IN EBR-II
Be/Cu (Braze)	-	4
Be/Cu (ps)	2	-
SiC/C (CVD)	-	2
TiC/C (CVD)	4	4
TiB_2/C (CVD)	2	2
Mo/Cu (ps)	4	-
Mo/V/Cu (ps)	1	-
TiC/Mo (ps)	1	-
Be/304SS (ps)	2	-
TiB_2/304SS (ps)	1	-
V/TZM **(eb)**	2	-
TiC	2	-
TiB_2	2	-
TiB_2 + BN	2	-
SiC	4	-
SiC + (B+C)	2	-
	31	12
	2-3 DPA	6-12 DPA
	~ 450°C	~ 450°C

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