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Damage Analysis and Fundamental Studies

Proceedings of a Workshop on Fusion Environment Sensitive Flow and Fracture Processes

February 1981

U.S. Department of Energy
Assistant Secretary for Energy Technology
Office of Fusion Energy
Washington, DC 20545

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FUSION ENVIRONMENT SENSITIVE FLOW AND FRACTURE PROCESSES

A Workshop held on
August 4-5, 1980
Battelle Seattle Research Center
Seattle, Washington

Organized by:

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Battelle Memorial Institute
Richland, Washington 99352

Chairman of "Fundamental Mechanical Behavior Subtask Group"

Damage Analysis and Fundamental Studies Task Group

PREFACE

This is the proceedings of a workshop on "Fusion Environment Sensitive Flow and Fracture Processes," held in Seattle on August 4-5, 1980. The workshop was a planning activity of the Subtask Group on Fundamental Mechanical Behavior of the Damage Analysis and Fundamental Studies Task Group. In order to give the recommendations formulated at the workshop wide distribution, the proceedings are issued here as a companion volume to the regular DAFS Program Quarterly Progress Report.

CONTENTS

I. SUMMARY 1

II. WORKSHOP APPROACH 5

III. WORKSHOP SCHEDULE 7

IV. WORKING GROUP PARTICIPANTS 8

V. MATERIALS RESEARCH NEEDS 9

VI. WORKING GROUP RECOMMENDATIONS 14

A. FLOW PROCESSES AND PROPERTIES 14

B. EFFECT OF FLOW PROCESSES ON FRACTURE 22

C. TIME DEPENDENT FRACTURE 32

D. RADIATION INDUCED OR ENHANCED EMBRITTLEMENT 43

I. SUMMARY

The Damage Analysis and Fundamental Studies (DAFS) Task Group was recently reorganized with the establishment of a subtask group on "Fundamental Mechanical Behavior". While there are several programs in the DAFS program addressing mechanical behavior issues, there has been no recent discussion of the mechanical behavior issues which are relevant to the overall fusion materials program. Therefore, this workshop was organized as a planning activity for the "Fundamental Mechanical Behavior" subtask group and to introduce researchers in the DAFS programs to the relevance of various mechanical behavior issues. Participants were chosen to represent both subtask B* and C* of DAFS and subtasks of the Alloy Development for Irradiation Performance task group.

The workshop was divided into working groups with each group responsible for listing research activities which they considered necessary to understand, interpret and control the mechanical properties of fusion reactor materials. These activities were to include a balanced blend of theory, model development and experiment.

The recommended research activities are summarized below in order of decreasing priority within a working group. 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 for evaluating 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.

A. FLOW PROCESSES AND PROPERTIES

1. Establish correlations of radiation hardening with microstructure (high priority)
2. Investigate the effect of a cyclic flux on creep rate (high priority)
3. Investigate the flux and temperature dependence of radiation-enhanced creep at temperatures below where fission data are available (high priority)
4. Investigate the effect of temperature cycling on creep rate (medium priority)
5. Measure the effects of helium on radiation creep rate (medium priority)
6. Investigate the effect of stress cycling on creep hardening and microstructural evolution (medium priority)
7. Investigate the nature of deformation during synchronized temperature, flux and stress cycles (medium priority for modeling, low priority for experiment until modeling completed)

B. EFFECT OF FLOW PROCESSES ON FRACTURE

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 basic experimental studies to identify the microstructural causes and conditions for channel deformation (high priority)
4. Evaluate ductile and brittle fracture models (medium priority)
5. Evaluate a standard Lifetime Analysis Code for the first wall of a magnetic fusion reactor (low priority)

C. TIME-DEPENDENT FRACTURE

1. Evaluate the need for near threshold crack propagation evaluation for fusion design (high priority)

2. Provide liaison with other task groups on stress corrosion and corrosion fatigue (high priority)

3. Develop models for creep-rupture

   a. Flag ADIP creep rupture specimens for microstructure archives (high priority)

   b. Calibrated correlation model (CCM) development (medium priority)

   c. Alternate pressurized tube sample design (medium priority)

   d. Microstructural analysis of creep rupture specimens (low priority)

4. Evaluate the relevant creep-fatigue interactions in fusion reactor first wall designs (medium priority)

0. RADIATION-INDUCED OR -ENHANCED EMBRITTLEMENT

1. Austenitic stainless steels

   a. Evaluate the role of helium on ductility at low temperatures (high priority)

   b. Develop property-property correlations (high priority)

   c. Develop techniques for measuring grain boundary helium and impurities (medium priority)

   d. Write a summary report on hydrogen embrittlement of austenitic stainless steels (low priority)

2. Ferritic steels

   a. Evaluate the role of trace impurities on embrittlement (high priority)

b. Develop property-property correlations (high priority)

c. Evaluate saturation in ΔDBTT of pressure vessel steels and its relevance to fusion materials (medium priority)

d. Evaluate the potential for hydrogen embrittlement of irradiated materials (low priority)

II. WORKSHOP APPROACH

As a planning activity, the objectives of the workshop were to list, prioritize and milestone the activities necessary to understand, interpret and control the mechanical behavior of candidate fusion reactor alloys. Emphasis was placed on flow and fracture processes which are unique to the fusion environment since the national fusion materials program must evaluate these effects without assistance from other reactor programs. The working group on flow processes and properties was concerned with the time-dependent and independent flow. This included radiation hardening and steady state and cyclic creep. The working group on the effect of flow on fracture was concerned with the relationships between the unique flow properties of irradiated materials, such as dislocation channeling, and the fracture properties of these materials. The working group on time-dependent fracture was concerned with high-temperature, time-dependent fracture, such as stress-rupture and fatigue fracture, while also being concerned with time-dependent environmental effects on fracture. The working group on radiation-induced or enhanced embrittlement was concerned primarily with time-independent fracture of materials for near-term fusion device applications. Therefore, this group was concerned primarily with austenitic and ferritic stainless steels. The participants in each of these working groups are listed in section IV.

A presentation by R. E. Nygren on material needs for ETF was also included in the workshop schedule, section III, to assure the most recent input on this subject. Long-term material needs are considered equally important to near-term needs, but it was assumed that the participants were familiar with the long-term research needs.

The workshop schedule, section III, was structured so that most of the first day was devoted to the presentation of summaries and recommendations by each working group chairman to all the workshop participants. Comments and discussion of the recommendations were encouraged during this time. The working groups met separately for most of the second day, with a wrap-up session at which the chairmen presented an up-dated version of the

recommendations. Each chairman was then responsible for a written summary of the recommendations and the rationale for their selection, while the workshop organizer was responsible for the final report.

III. WORKSHOP SCHEDULE

SUNDAY - AUGUST 3

7:00 pm Reception

MONDAY - AUGUST 4

8:00 Opening
8:15 Material Needs for ETF: R. E. Nygren
8:45 Radiation-Induced or-Enhanced Embrittlement: R. H. Jones
10:15 Break
10:30 Flow Processes: A. P. L. Turner
12:00 Lunch
12:45 Effect of Flow Processes on Fracture: W. G. Wolfer
2:15 Break
2:30 High Temperature Fracture: G. Lucas
4:00 Working Groups
6:00 Depart for Tillicum Village for Dinner

TUESDAY - AUGUST 5

8:00 Working Groups
12:00 Lunch
1:00 Working Groups
3:00 Workshop Summaries (30 min. per group)
5:00 Workshop Wrap-Up

IV. WORKING GROUP PARTICIPANTS

1. Flow Processes	*1. A. P. L. Turner, ANL 2. W. Barmore, LLL 3. E. R. Bradley, PNL 4. F. Garner, HEDL 5. L. K. Mansur, ORNL
2. Effect of Flow Processes on Fracture	*1. W. G. Wolfer, U. Wisc. 2. W. Jesser, U. Virginia 3. M. Grossbeck, ORNL 4. J. L. Straalsund, HEDL
3. High Temperature Fracture	*1. G. Lucas, U.C.S.B. 2. G. R. Odette, U.C.S.B. 3. E. R. Gilbert, HEDL
4. Radiation-Induced or -Enhanced Embrittlement	*1. R. H. Jones, PNL 2. R. E. Nygren, ORNL/HEDL 3. G. L. Wire, HEDL 4. J. Spitznagel, WRD 5. S. Rosenwasser, G.A.

Others: D. G. Doran, HEDL

*Chairman

V. MATERIALS RESEARCH NEEDS - W. G. Wolfer, U. of Wisc.

The effect of radiation damage, helium and hydrogen embrittlement, and surface erosion are major considerations with regard to the ultimate lifetime of the first wall in a fusion reactor. All the above factors are expected to significantly affect the mechanical and fracture properties of candidate materials for the first wall.

Experimental investigations of the above phenomena which would be both relevant and timely to the design of fusion reactors are hampered by the lack of a fusion irradiation facility. Instead, for the foreseeable future, only fission reactors and 14 MeV neutron sources are available for irradiation studies to simulate the fusion environment. In order for these studies to be meaningful, it will be necessary to:

1. identify the materials properties which significantly influence the lifetime of the first wall;
2. provide the necessary understanding to correlate the mechanical property changes with the radiation environment;
3. find physical connections between materials properties relevant to design and those which can be studied and measured with available or soon to be available irradiation facilities.

Whereas the second task (2) is adequately considered in the DAFS program, the first and third tasks (1 and 3) require greater emphasis in the future. The reasons will be discussed briefly in the following two sections.

A. METHOD FOR IDENTIFICATION OF RELEVANT MATERIALS PROPERTY CHANGES

Relevant materials properties and their changes are those which significantly affect the lifetime of the first wall. Based on previous design studies, it is generally believed that both fatigue and creep crack growth, as well as unstable crack propagation, will limit the lifetime of the first wall. Excessive dimensional changes due to swelling and irradiation creep

can possibly by avoided by judicious design and proper selection of materials and operating temperatures. Nevertheless, swelling is not insignificant. Differential swelling, caused by the large temperature gradient through the first wall, may produce large residual stresses even at low average values of swelling. These residual stresses can accelerate the crack growth and thereby limit the lifetime.

This is but one of the many synergistic effects or connections that need to be uncovered and explored in view of their impact on fusion reactor design. Failure to identify these connections may result in an extraneous materials research and development program.

A more systematic approach to cover connections and focus the research on the important problems must emerge from a closer interaction between fusion reactor design and materials development. This interaction not only involves better comnunication and collaboration between designers and materials scientist, but also the development of a framework to assess and quantify the materials performance in the context of the first wall integrity. Design procedures adopted so far have been incomplete, inconsistent, or simply too narrow to provide the comprehensive methodology needed.

Nevertheless, as a result of all the previous design studies, a certain systematic picture has emerged. Although this picture lacks the necessary details, it clearly indicates that a generic methodology can and should be developed to provide the capability for lifetime predictions and quantitative materials assessment. This methodology, in the form of models and computer codes for a generic first wall section, can then be utilized to test materials property correlations and to guide further materials data development.

B. TYPICAL EXPOSURE OF THE FIRST WALL MATERIAL

To appreciate the significance of mechanical property changes we describe briefly the magnitude of the stresses in the first wall.

Suppose the first wall consists of a type 316 stainless steel shell structure of 0.5 cm thickness. The steady-state heat flux during the plasma burn is assumed to be 50 W/cm^2 at about 2 MW/m^2 wall loading. For a shell constrained from bending, the thermal stresses are ± 340 MPa on the surfaces. Although these thermal stresses relax eventually during the plasma burn because of irradiation creep, they give rise to equal and opposite residual stresses during the plasma-off periods.

The stresses due to the coolant pressure are expected to be between 10 to 100 times the coolant pressure, depending on the design. Accordingly, they can be kept at acceptable levels by proper design of the first wall and the blanket.

The magnetic loads induced during a plasma disruption may generate equivalent pressures of about 1.4 MPa. Hence, the associated membrane stresses may reach values up to 135 MPa. Again, as in the case of the coolant pressure, the stresses produced by the magnetic loads can be minimized by judicious design.

Much more serious than the magnetic loads are the transient thermal stresses produced in the first wall during a plasma dump. The heat flux may reach values in excess of 10 kW/cm^2 during a time on the order of 20 ms. As a result, the surface temperature of the first wall rises beyond the melting point. A layer of about 40 μm thickness may melt, of which a substantial part may also evaporate. The heat **pulse** penetrating into the solid remainder of the first wall thickness produces transient thermal stresses. In front of the thermal pulse, the stresses are tensile and on the order of 270 MPa, whereas under the heat pulse the compressive stresses yield the material. During the cool-off period following the plasma dump, reversed plastic deformations take place. In all, up to 20% of the wall thickness may be subject to either unidirectional and/or reversed yielding. If sudden plasma dumps cannot be ruled out entirely, a minimum ductility value of at least 0.2% must be required, over the lifetime of the first wall. Unfortunately, embrittlement caused by radiation damage and helium transmutations may degrade the ductility to lower values.

In addition to these presumably rare events of extreme cyclic loading, the wall may be subject to normal cyclic stresses when plasma operation is not continuous. Part of the stress cycle is formed by the residual thermal stresses during the off-period, and part arises from the stresses produced by differential swelling. Their magnitude is in proportion to the temperature difference across the first wall and to the ratio of the swelling and irradiation creep rates. Based on the present correlations for swelling and irradiation creep of 316 SS, the stresses may reach values of ± 540 MPa for an average wall temperature of 350°C, i.e., for a temperature variation from 375°C to 325°C across the wall. For a cyclic plasma operation, the surface stresses on the first wall fluctuate between ± 540 MPa and ± 240 MPa, where the plus (minus) sign holds for the surface facing the coolant (plasma).

Based on this brief discussion of the loading of the first wall, the most relevant flow and fracture properties are:

1. Stage I and Stage II of fatigue crack propagation for normal cyclic plasma operation;
2. Stage III of fatigue crack propagation and fracture toughness K_{IC} for abnormal plasma operation;
3. Yield stress and ductility under transient temperature and stresses occurring during a plasma dump.

Crack growth through the first wall must be studied not only in simple tension tests but also in steep stress gradients and under compressive loads. The complicated stress variation both with time and through the wall thickness will require the development of analytical crack growth models and codes which must be calibrated against actual crack growth data from tests under simpler stress histories and stress distributions.

It will not be possible for a long time to investigate fatigue crack growth, ductility loss, differential swelling, and irradiation creep in a real fusion environment. Furthermore, irradiation experiments in either

fission reactors or 14 MeV facilities will be mostly restricted to small samples. It is, therefore, mandatory that properties measured on these small samples can be related to the more macroscopic properties involved in plastic flow and fracture. Microstructural, flow, and fracture properties require samples of increasing size. Therefore, connections between microstructure and flow, and between flow and fracture, must be sought and established.

VI. WORKING GROUP RECOMMENDATIONS

A. FLOW PROCESSES AND PROPERTIES, Chairman: A. P. L. Turner, ANL

1. Introduction

The subgroup on flow processes and properties focussed its attention on irradiation-enhanced creep and radiation hardening. The time spent on the former greatly exceeded that spent on the latter primarily because we felt that there are many more unknown factors in radiation-enhanced creep than there are in radiation hardening. We did feel that flow stress measurements and correlations of flow stress with microstructural characterization form the basis for attempts to relate other mechanical properties to microstructure. Therefore, we have recommended that such work should continue to receive a high priority.

In the area of irradiation creep we have tried to emphasize questions that relate to the unique aspects of the fusion reactor environment. Therefore, the effects of cyclic flux, temperature and stress are prominent in our recommendations. The effect of He generation is also included, but the effect of He on flow is probably less critical than its potential effects on fracture. This is reflected in our priorities. We also identified behavior at low temperature as being an area of relative ignorance that will be important to near-term devices. Although some of the most pronounced effects of irradiation on flow relate to changes in the deformation mode to more localized shear, we did not include this effect in our discussions because we expected it to be covered by the group considering the effects of flow on fracture. We confined our discussions to questions that were related to changes in flow strength or deformation rate resulting from irradiation.

In setting priorities, we attempted to be realistic about what can be achieved. Some questions that may be very critical received a low priority because it is not clear that they can be answered without a major

effort. They should, therefore, wait until easier preliminary experiments give a surer indication of their importance. The items given high priority are areas where knowing the answer is important for material selection and reactor design, where little or no information exists and where it is realistic to expect that an effort commensurate with current and anticipated funding levels will produce results. Most of the recommendations relate to experiments. This is not meant to exclude theory. A reasonable amount of theory and modelling will be required in all of the efforts mentioned in order to interpret the experimental results and assess their generality. It was felt that theories of radiation-enhanced creep need to be improved, but no recommendation was made that this be a high priority. Partly this is because such theory development is being done outside the DAFS program and partly because more extensive experimental results are needed before theories can realistically be advanced.

In the recommendations, we have not been specific about what materials should be used in experiments. In some cases, it will be most appropriate to use the engineering alloys that are prime candidates for use in the first wall. In other cases, there may be excellent reasons to use model alloys. The choice would depend on the objective of the specific experiment. In most cases, working with pure materials would be less informative than working with alloys because solute-point-defect interactions and segregation will always play an important role in real alloy systems.

2. Recommendations

a. Establish Correlations of Radiation Hardening with Microstructure (High Priority)

Correlation of measured values of the flow stress with values predicted from the microstructure is the best available test that microstructure characterizations are complete and accurate. The relationship between microstructure and flow stress is probably the best-understood connection between structure and a mechanical property. The understanding of this

relationship will be an important ingredient in developing the other structure/mechanical property relations that will be one basis for extrapolating test data to fusion reactor conditions. Therefore, the work on correlating flow stress with microstructure should continue to have a high priority in the mechanical behavior efforts. This will refine our understanding of what are the important characteristics of microstructure for determining mechanical behavior. It should also be noted that at low irradiation temperatures where TEM is not useful for characterizing the microstructure, the flow stress may be the most suitable quantitative measure of radiation damage.

This is an ongoing effort involving both Subtask Groups B and C. It should continue to receive high priority

b. Investigate the Effect of a Cyclic Radiation Flux on the Acceleration of Creep Deformation (High Priority)

The cyclic nature of the radiation flux is one of the ways that the fusion environment differs from that of fission reactors. Consequently, there is only very sparse experimental evidence available regarding the effect of cyclic irradiation on creep, but what evidence is available generally shows that a cyclic flux accelerates irradiation creep. The theoretical basis for believing that cyclic irradiation should accelerate creep is that excess point defects of a single type should arrive at sinks during the transient periods following an increase or decrease in the damage rate. The dislocation climb caused by the net flux of interstitials or vacancies can release the dislocations from obstacles allowing strain to be generated by dislocation glide. Since the glide strain is in the direction of the stress, independent of the climb direction, the strain increments from both positive and negative transients are additive. Theoretical estimates of the size of the effect are very dependent on the details of the model used. They can predict either no effect or a large effect, depending on what assumptions are made about the climb distance required to release a dislocation and the strain per release. Choices of these parameters cannot be made reliably, so an experimental investigation of the effect is required.

Cyclic flux experiments can best be done using light-ion irradiation from an accelerator. Damage rates in such an experiment are comparable to those projected for a fusion reactor and the flux per pulse (in terms of dpa), temperature and stress levels can easily be chosen to cover the range appropriate to the reactor. Such experiments are necessarily limited to a small total dose. This should be no disadvantage in determining the affect of flux cycling on the creep rate at a given microstructure, but does not tell how the cyclic flux might alter the microstructural evolution from that observed during steady state irradiation. The experiments should be done on alloys of interest for reactor applications or suitable model alloys. Model alloys which are more stable than, for example, austenitic stainless steel may be more appropriate for initial experiments designed to assess the magnitude of the effect independent of phase changes.

c. Investigate the Flux and Temperature Dependence of Radiation Enhanced Creep at Temperatures Below Those Where Fission Reactor Data are Available (High Priority)

Very few experimental data on radiation creep rates at low temperature are available. It is conventionally believed that the creep rate should drop off with decreasing temperature below the peak swelling temperature in much the same way as swelling does. Some experimental evidence indicates that this may not be the case and that the creep rate may even increase at low temperatures. A similar situation exists with respect to the dependence of creep rate on damage rate. Although it is recognized that the radiation creep-rate should increase somewhat less than linearly with flux or damage rate because recombination of point defects becomes more important at high damage rate, some experimental evidence exists to indicate that the creep rate:flux ratio rises more rapidly at low flux than anticipated. Because near-term fusion reactors will operate at temperatures well below those of fission reactors, these effects are of concern.

The investigation of these effects should include both a more thorough search for existing experimental evidence from the various reactor programs

and new experiments. Additional data may be available to support the idea that anomalous behavior occurs in the low stress and low flux regimes. A comprehensive survey of all available sources and re-evaluation of old data should be carried out. Experiments using accelerator sources should be extended to lower temperatures. If a change of mechanism occurs at low temperature to account for abnormally high creep rates, this should be detected by such experiments. Low temperature, high dose reactor irradiations of pressurized tube samples should be carried out to determine whether some aspect of microstructure evolution at low temperature can produce greater than expected creep rates. (Some pressurized tube samples have been included in ADIP irradiations. DAFS should examine their planned test matrix to see if additional samples should be included to cover the low temperature regime.) Experiments should be designed to cover the widest possible flux range in order to provide a basis for correlating results from all irradiation facilities.

This regime is of prime concern for near-term devices and should receive high priority.

d. Investigate the Effect of Temperature Cycling on the Acceleration of Creep Deformation (Medium Priority)

The effects of cyclic temperature changes on radiation-enhanced creep during constant flux irradiation are expected to be similar to those produced by a cyclic flux at constant temperature. The temperature changes alter the point defect mobilities and cause pulses of interstitials or vacancies to arrive at sinks as the point defect concentrations adjust to the new conditions. If a cyclic radiation flux is found to have a major effect on creep rate, experiments should be extended to investigate the cyclic temperature effects because both types of cycles will be present in a fusion reactor. An additional reason for investigating temperature cycles is that temperature cycling is possible to do in-reactor whereas flux cycling is not. If a good correlation can be established between the

effects of temperature cycles and flux cycles, cyclic temperature experiments in a reactor offer a means to carry the investigation of such effects to high dose levels where the influence of cycling on creep through alteration of the microstructural evolution can be evaluated.

The priority for this work should be increased if cyclic flux conditions are found to have a major effect on creep.

e. Measure the Effects of He on the Radiation Creep Rate (Medium Priority)

Because steady-state irradiation creep depends on the bias for dislocations absorbing net point defects of one type, it can be expected to be affected by any changes in sink structure or defect mobilities. The high concentrations of gas generated in materials by the high energy fusion neutrons can modify irradiation creep in two ways: by trapping the point defects and thereby modifying their mobility and the defect currents to sinks, and by altering the nature of the radiation-produced microstructure. Effects of defect trapping can be investigated using light-ion irradiations of He-injected samples. The effects of He generation on microstructure development are being investigated using mixed spectrum reactor irradiations and dual-ion irradiations. Correlations of microstructural changes with creep rate changes can be obtained from creep measurements on samples pre-irradiated in reactor. A direct comparison of creep with He vs. creep without He can be obtained from irradiations of pressurized tubes of Ni-bearing alloys with some samples shielded from the thermal neutron flux to suppress He generation by the two-step reaction.

Comment by L. Mansur, ORNL:

The effects of high levels of helium on irradiation creep rate may be quite important where fine dispersion of He-vacancy clusters provide obstacles to dislocation motion. I would, therefore, recommend that the effects on irradiation creep of high helium levels be viewed as a high priority area. High rates of He generation in low dose ion irradiation

experiments would not suffice to examine this effect. Preinjection of specimens or preliminary reactor irradiation would hold promise for obtaining some similarity to the fusion reactor case.

f. Investigate the Effect of Stress Cycling on Creep, Hardening and Microstructural Evolution (Medium Priority)

The thermal-stress levels calculated for the first wall of a fusion reactor in recent design studies, including the ETF design, are sufficiently high that some cyclic microplasticity is to be expected. It is known that, in unirradiated material, cyclic plasticity, even at very small plastic strain amplitudes, can cause large changes in the dislocation microstructure. Cyclic plasticity causes cyclic hardening in annealed materials and cyclic softening in work-hardened materials. The cyclic deformation may eventually lead to the initiation of a fatigue crack and this aspect of the cyclic stress environment is discussed elsewhere. Even if a fatigue crack is not formed, or if it will not propagate, the cyclic plasticity can have a pronounced effect on the material mechanical behavior.

Cyclic microplasticity may change the level of radiation hardening achieved in the sample. Moving dislocations can be much more effective sinks for point defects than stationary ones because they will sweep up the point defects in their path. In addition, the cutting of radiation-produced dislocation loops by the moving dislocations can keep the loops from growing or destroy them completely. Dislocation generation during cyclic plastic deformation may keep the dislocation density in cold-worked material from decreasing during irradiation and thereby alter the microstructural evolution sequence. Cyclic deformation often increases the rate of creep deformation in unirradiated material.

The effects of cyclic stressing and resulting cyclic microplasticity discussed above will only be observed when the stress cycling is done simultaneously with the irradiation. Postirradiation fatigue tests will not reveal any effects on microstructural evolution. In postirradiation

tests, microplasticity will be suppressed by radiation hardening which will not have occurred during the initial portion of the life of a fusion reactor first wall. If cyclic deformation prevents or reduces the amount of radiation hardening, postirradiation tests may be made on material that is stronger than the first wall material will ever become. For these reasons, investigation of the effects of cyclic stressing must be done by in-situ experiments. Because the effects of most concern are associated with changing the evolution of the microstructure, the experiments must be carried to reasonably high dose levels. Because samples of a size suitable for mechanical testing must be uniformly damaged, the experiments will have to be done in a reactor. Although this is a difficult experimental problem, it should be possible to do such experiments in the FFTF.

g. Investigate the Nature of Deformation During Synchronized Temperature, Flux, Stress Cycles (Medium Priority)

The largest stresses in the first wall of a fusion reactor arise from temperature gradients and/or swelling gradients. Consequently, the net force through the wall is quite small and it is generally assumed that gross distortion from plastic deformation and creep will not occur. However, in the reactor, the cycles of flux, temperature and stress are synchronized so that, for example, at some points in the wall maximum compression stress always occurs at maximum temperature and flux while maximum tensile stress occurs at minimum temperature with no radiation flux. Under these conditions, it is possible for a "ratcheting" effect to occur so that even when the stress averaged over the wall thickness or averaged in time is small, a large net deformation in one direction can occur.

The possibility that large deformations can be accumulated by ratcheting during the fusion reactor stress, temperature, radiation cycles should be investigated by suitable theoretical models. Although there is some uncertainty about material deformation behavior in the fusion reactor environment, constitutive relations for deformation that are qualitatively correct can be constructed from available information. This should be

sufficient to determine if any unusual behavior is to be expected from the synchronized cycle. Improvements in the constitutive laws for deformation can be included in the models as additional experimental data from experiments on flux or temperature cycling become available. Experimental efforts to simulate the complicated first wall cycle should be undertaken only if the models predict a substantial effect from the combined cycle.

The theoretical modeling of the first wall response to its complex cycle should receive medium priority. Experimental simulation of the stress, temperature flux cycle should have low priority unless models indicate behavior that is critical to design.

Comment by L. Mansur, ORNL:

Synchronized temperature, flux and stress cycles may lead to unexpected effects in complex alloys because of the known great sensitivity of phase instability to temperature und dose rate (and possibly stress?). In pure metals, where only point defect kinetics need be considered, the compound effect of the three types of cycles might be anticipated from knowledge of the separate effects. My recommendation would he that the compounding effects of the three types of cycles receive high priority for technological alloys.

B. EFFECT OF FLOW PROCESSES ON FRACTURE, Chairman: W. G. Wolfer, U. of Wise.

1. Introduction

Relationships between microstructural and flow properties are discussed in the previous section. Accordingly, we emphasize in this section the connection between flow and fracture properties. The following discussion is divided into three parts:

- Models for fracture toughness for continuous deformations;
- Models for fracture toughness of materials with flow localization; and
- Models for fatigue crack growth.

a. Fracture Toughness for Continuous Deformation

Models which fall into this category are based on the assumption that plastic deformation ahead of the crack tip is governed by the same flow law that is being measured on a macroscopic tensile sample. As a result of this assumption, the plastic deformation is continuously (but not uniformly) distributed around the crack tip. The fracture toughness can then be described in terms of three flow parameters: the yield stress, the uniform or the ultimate strain, and the strain-hardening exponent. Furthermore, the fracture toughness depends also on at least one microstructural parameter, such as the dimension of the plastic zone or the spacing between brittle precipitate particles.

As an example of the predictive capability of two of these models, Figure 1 shows the predicted fracture toughness K_{IC} for HFIR irradiated stainless steel [1]. The two models chosen [2,3], the Hahn-Rosenfield [2] model (HR) and the Krafft [3] model (K), give substantially different values of K_{IC}. A factor of two appears to be the typical uncertainty for these types of models, and it originates from the ill-defined microstructural parameter contained in the model.

A better definition of the relevant microstructural parameters as well as a calibration of the models with K_{IC} data from irradiated samples appears to be the most fruitful way to improve the predictive capability of present models for fracture toughness.

The HR and K models apply to ductile failure. With regard to brittle fracture of bcc materials below the ductile-to-brittle transition temperature (DBTT), the model developed by Ritchie, Knott, and Rice (RKR) [4] appears to be the most promising. It relates K_{IC} to the yield stress, the strain hardening exponent, and two microstructural parameters. The latter are the cleavage stress and a length scale which is thought to be the average distance between brittle grain boundary phases in ferritic steels. In applying the RKR model to nuclear pressure vessel steels, it is usually

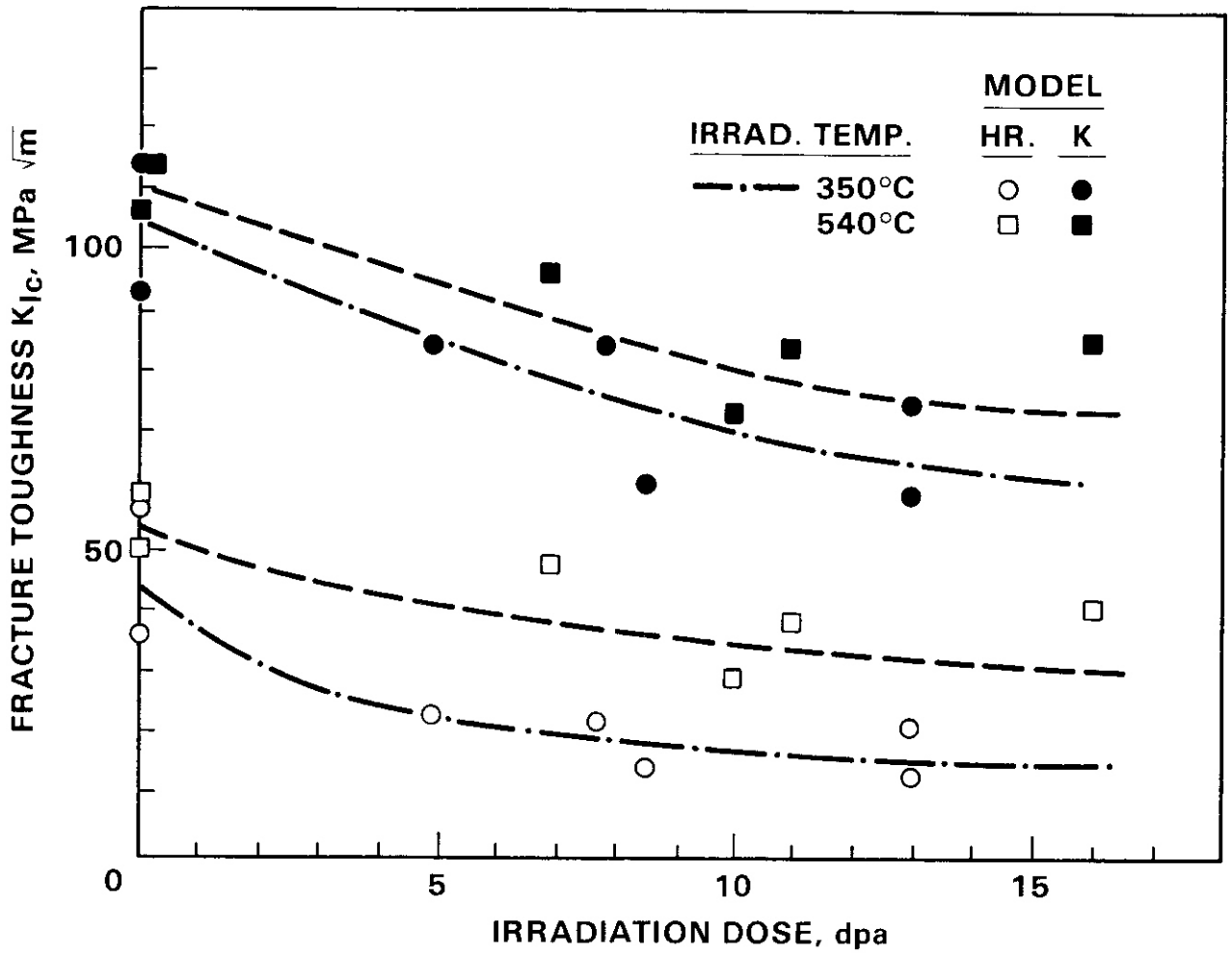

FIGURE 1. Fracture Toughness, K_{IC}, Versus Irradiation Dose for HFIR Irradiated Stainless Steel Predicted by the Hahn-Rosenfield (2) and Krafft (3) Models.

assumed that the two microstructural parameters are not affected by the irradiation. Although this assumption seems to work for pressure vessel steels, it may not hold at high irradiation doses and increasing gas production in a fusion reactor first wall.

b. Fracture Toughness for Localized Flow

High-strength materials and irradiated metals often exhibit localized plastic flow. The macroscopically-measured flow properties then bear little resemblance to the microscopic shear in the localized deformation channels. Since localized flow occurs also at crack tips, the stress concentration is no longer related to the macroscopic flow properties as assumed in the HR, K or similar models.

The understanding of both the localized flow phenomenon and its effect on fracture toughness is in an embryonic state. Its importance, however, to flow and fracture of irradiated materials demands a strong research effort. The following discussion summarizes briefly what is known about this flow localization.

1) Conditions for Flow Localization

Flow localization is thought to occur when the rate of work-softening exceeds the rate of work-hardening. Quantitative models require the knowledge of the specific causes for work-hardening and work-softening. As a typical example, consider the model of Mori and Mura [5] developed for dispersion-hardened alloys. Work-hardening is produced by the back-stress of the nondeforming inclusions, whereas work-softening is due to the annihilation of shear dislocation loops after they contributed a critical value of shear to the deformation in the channel. Depending on the value of this critical shear parameter, as well as the density of loops and inclusions, the plastic flow distribution can change from homogeneous to localized and back to homogeneous as the macroscopic strain increases.

In contrast to the microstructural modelling of localized flow, Rice [6] has recently developed a phenomenological approach. He derived forms of constitutive deformation laws which lead to flow localization when certain empirical parameters assume critical values. A microstructural interpretation of these parameters is possible, though not necessary, in terms of dislocation cross slip, ductile hole growth, etc.

Experimental observations of flow localization in irradiated materials have been made in a few materials [7]. Presumably, this phenomenon is not restricted to a few peculiar cases, but is of rather general nature. Presently, the observations fall into two groups. First, flow localization is seen in materials irradiated to low doses and at low temperatures, and is connected with the sweeping-up of small defect clusters by glide dislocations.

Second, materials irradiated to high doses and at temperatures where swelling occurs exhibit the so-called channel deformation indicated by seared voids. It is not clear at the present time whether the voids are instrumental or only conspicuous indicators in the flow localization process. Observations of flow localization in porous materials would indicate, however, that radiation-induced voids do play a significant role in channel deformation.

Accordingly, investigations of the conditions for flow localization in irradiated materials is strongly recommended because of its impact on the ductility and the fracture behavior of first wall candidate materials.

2) Relation Between Flow Localization and Crack Propagation

Although it is well known from experiments on high-strength materials that flow localization and low ductility are related, only circumstantial evidence exists that other fracture properties, such as fracture toughness and fatigue crack growth, are similarly degraded.

High strength alloys have the propensity for flow localization, low ductility and fracture toughness. Furthermore, flow localization clearly favors fatigue crack initiation and growth in Stage I. Materials which acquired their strength by radiation damage are expected to follow this trend.

Smith, Cook, and Rau [8] have recently developed a more mechanistic basis for these observed trends. They relate the fracture toughness to the yield strength and a critical decohesion strain for the flow channel. Unfortunately, this critical decohesion strain lacks any quantitative basis in terms of other measurable quantities, and their model can, therefore, supply only qualitative answers at the present time.

In the observed deformation channels of irradiated stainless steels, it is possible to measure the accumulated shear from the shape of the sheared voids. Assuming that the critical decohesion shear is at least as large as this measured shear, a lower estimate of the fracture toughness can be made with a modified model of the original one by Smith et al. [8]. By this approach, it is estimated that the fracture toughness of highly irradiated 304 SS duct material is no lower than 27 MPa $\sqrt{m}$.

It appears then, that the fracture toughness of this material is still satisfactory even though the ductility, as measured in a tensile test, is practically zero.

c. Relationships Between Fracture Toughness and Fatigue Crack Growth

Stage III of fatigue crack growth is determined by the value of the static fracture toughness K_{IC}. A reduction of the latter due to radiation embrittlement will narrow the stress range for Stage II also. Furthermore, based on empirical correlations between fracture toughness and the exponent m in the Stage II crack growth law

$$da/dN = C(\Delta K)^m,$$

the exponent m increases sharply as K_{IC} decreases; this correlation is shown in Figure 2.

Hence, the following qualitative connections are expected also for radiation-hardened materials: with increasing radiation damage K_{IC} drops, as a result, all stages of fatigue crack growth move to lower ranges of ΔK; furthermore, the stress range for Stage II narrows, thereby increasing the exponent m.

2. Recommendations

Ductility, fracture toughness, and fatigue and creep crack growth have been identified as crucial mechanical properties for a fusion reactor first wall. All these properties are subject to changes because of radiation damage, helium and hydrogen embrittlement, and temperature excursions. Even if these properties could be measured in available irradiation facilities and with standard test procedures, there would remain questions as to

- their applicability to the fusion radiation environment,
- and their applicability and relevance to the complex stress and temperature history in a fusion reactor first wall.

In order to ensure a timely resolution of these questions, the following recommendations are made. These recommendations may overlap with some of the objectives of other tasks or subtasks in the fusion materials program, and therefore they may not be germane only to the Fundamental Mechanical Behavior Subtask of the DAFS program. Nevertheless, these recommendations are made from the viewpoint of the general objective of the DAFS task and, in particular, the Fundamental Mechancial Behavior Subtask, namely to gain a basic understanding of the material issues and the materials performance in the first wall and blanket structure of a future magnetic fusion reactor.

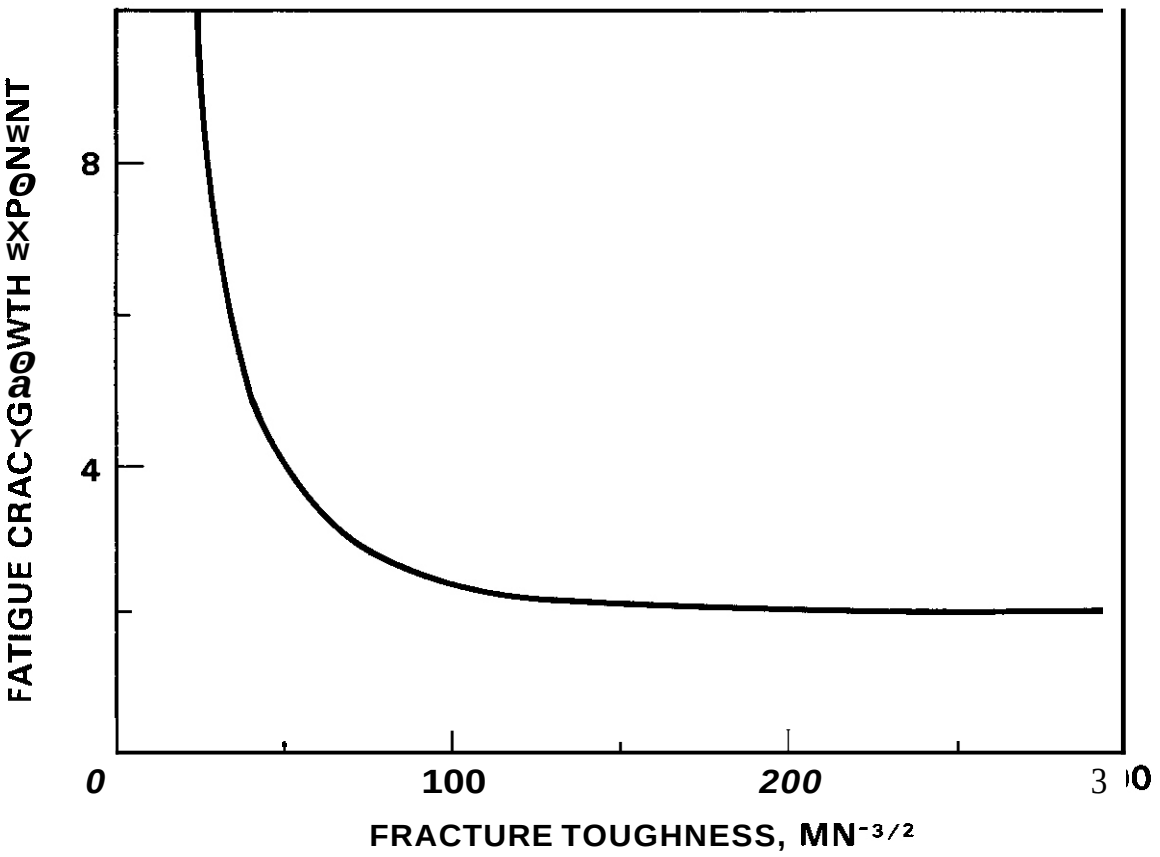

FIGURE 2. Fatigue Crack Growth Exponent m Versus Fracture Toughness.

a. Perform Fracture Toughness Experiments on Specimens of Highly Irradiated Duct Material (High Priority)

This material is available and is known to exhibit channel deformation. The purpose of these experiments is to clarify the relationship between channel deformation, lack of ductility as indicated by the absence of a macroscopic fracture strain, and fracture toughness. If these experiments were to result in an invalid fracture toughness test because of the small specimen size, it would indicate a satisfactory toughness in spite of the ductility loss. Conversely, a valid test would provide important data for the fracture resistance of extremely radiation-hardened materials.

b. Perform Fatigue Crack Growth Experiments on Specimens of Highly Irradiated Duct Material (High Priority)

These experiments will have to be performed anyway in conjunction with the fracture toughness experiments. Their purpose is to identify the role of channel deformation in fatigue crack initiation and growth.

c. Initiate Theoretical and Basic Experimental Studies to Identify the Microstructural Causes and Conditions for Channel Deformation (High Priority)

HVEM studies are recommended as the experimental tool to explore the mechanism of flow localization and its effect on crack propagation.

d. Evaluate Ductile and Brittle Fracture Models (Medium Priority)

Models for ductile and cleavage fracture toughness should be refined to provide a more reliable basis for relating tensile properties to fracture properties. Close collaboration with the ADIP experimental program is recommended.

e. Develop a Standard Lifetime Analysis Code For The First Wall Of a Magnetic Fusion Reactor (Low Priority)

The purpose of this task is manifold: develop appropriate correlations for the mechanical properties and their interrelationships which are

compatible with fusion reactor design requirements; identify in a timely manner critical needs in the development of a data base for mechanical properties; develop mechanical design criteria for fusion reactor structural materials; and provide a tool for fusion reactor design which incorporates the results and contributions of the fusion materials development programs. This task is not only central to the efforts of the mechanical properties work in DAFS; but it establishes strong links to other fusion materials and design groups, in particular the Analysis and Evaluation group of ADIP, and the Correlation Methodology subgroup of DAFS.

The development of the Lifetime Analysis Code will at the same time ensure a closer interaction among various groups in the fusion materials program.

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Comments by J. A. Spitznagel, Westinghouse R&D Center:

1. *There is no mention of the possibility that flow localization can occur on a scale too small to be detected in the TEM. For example, it is likely that deformation channels occur in pressure vessel steels, yet they have never been seen because the defects responsible for the hardening are (for the most part) beneath the resolution limit of the TEM. How would one proceed with a ferritic alloy that exhibits zero tensile ductility but no visual evidence of channel fracture? It makes sense to proceed experimentally with the 304 SS and 316 SS material known to exhibit channel formation to evaluate effects on fracture toughness and da/dN, but the "nodeling and mechanisms" portion should get medium to low priority.*

2. *If a strong approach is taken to assessing "steady-state" effects when K_{IC}, da/dN, etc., may be more dependent on transient effects (plasma dump, etc.), we run the risk of appearing naive. Perhaps we should state that we are assessing a "simpler" pseudo-steady state and not the transient effects.*

C. TIME DEPENDENT FRACTURE, Chairman: G. Lucas, U.C.S.B.

1. Introduction

The basic mechanisms and the anticipated effects of high energy neutron irradiation and a fusion reactor environment on the operation of these mechanisms were considered for a number of time-dependent failure modes. These included creep rupture, fatigue, creep-fatigue, fatigue-environment interaction and stress corrosion cracking. As a result of discussions with the members of the general subtask group and with consultants to the working group on time dependent fracture, a near-term research plan was drafted and priorities were assigned to the proposed topics.

2. Recommendations

a. Evaluate the Need for Near-Threshold Crack Propagation Evaluation for Fusion Design (High Priority)

Historically, design against failure induced by cyclic loading (fatigue) has been largely based on the classical S-N curve, from which allowable applied stresses (S) may be chosen for a given life, or number of cycles to failure (N). However, recently there has been increasing use, in which safety is a prime consideration, of the defect-tolerant approach. In this approach, pre-existing cracks or defects must be assumed to exist, and component life is then limited by the time required to propagate the defects to a critical crack size.

Consequently, there has been considerable research in recent years to characterize crack growth rates da/dN as a function of an extensive array of test conditions, including alternating stress intensity AK, stress ratio R, temperature, environment, and metallurgical conditions. Unfortunately, much of these data have been generated in the intermediate and high crack growth rate regimes ($\geq 10^{-5}$ mm/cycle) - Stages II and III, respectively, where data are more easily obtained, but crack growth rates may be too high to be of technological importance, expecially for thin structures.

Techniques have been developed recently to ascertain small crack growth rates ($\leq 10^{-5}$ mm/cycle); and as a result, a number of important discoveries have been made in this low crack growth rate regime, often referred to as Stage I. For instance, for a given set of material and test conditions a threshold value of ΔK, ΔK_0, exists below which crack propagation is undetectable. As the applied alternating stress intensity approaches this value of ΔK_0, the functional dependence of da/dN on ΔK deviates significantly from the Paris law $da/dN = C(\Delta K)^m$; i.e., the Paris law can greatly overpredict crack growth rates in Stage I. Furthermore, the value of ΔK_0 is sensitive to both microstructure and test conditions,

such as R and the nature of the environment. For example, in most tests using compact tension specimens (long initial crack size, small applied stress $\Delta\sigma$), ΔK_0 decreases as the material strength or R increases. However, to be consistent with the known behavior of materials under high cycle fatigue conditions, it is anticipated that for small surface cracks or defects (small initial crack size, high no) that ΔK_0 will increase as material strength or R increases. Also, there are strong indications that environment plays a significant role in influencing the magnitude of ΔK_0 and da/dN in Stage I; generally, environments which promote static stress corrosion cracking aggravate crack growth in Stage I. Hence, of particular importance for near-term fusion device considerations are the environmental effects of hydrogen isotopes in ferritic steels and aqueous environments in austenitic stainless steels.

To date, emphasis in the fusion materials program has been placed on characterizing crack growth rates in the intermediate rate regime, and the MFE-3 and MFE-5 experiments are underway to address questions of irradiation effects on fatigue crack growth in this regime. However, of equal if not greater importance may be the characterization of Stage I behavior for the materials and conditions of relevance. In view of the relatively large number of anticipated stress cycles ($>10^5$), the thinness of the first wall structures, the environment, and the anticipated as-fabricated surface conditions of the structural components in fusion devices, the most important fatigue design consideration may be initiation/propagation of surface cracks under near-threshold crack propagation conditions. And the use of a Paris law extrapolated down to Stage I may lead to overly conservative designs.

Consequently, in view of the potential importance of Stage I crack growth behavior to the design of fusion systems, particularly near-term designs, and the relative scarcity of information on this failure mode as it pertains to fusion systems, the following research program is recommended in addition to the current research in this area as the highest priority item in the area of time-dependent failure:

As soon as possible, a combined topical report/recommended research program should be prepared to address the issue of near-threshold crack growth as it pertains to fusion device design, both for the near and long term. In the topical report section, emphasis should be placed on ascertaining the importance of Stage I crack growth in fusion reactor structures and identifying previous relevant research. The question of differences in long and short crack growth dependence on test and material conditions should be addressed. Environmental effects on Stage I crack growth should be addressed as well; specifically, the effects of H on ferritic steels and aqueous environments on austenitic stainless steels. The report should be researched and written with the aid of knowledgeable consultants (e.g., R. Ritchie, M.I.T. and L. James, HEDL); and the analysis of the relevance of Stage I crack growth to fusion device design should be performed with a comprehensive structural analysis technique, such as the WISECRACK code developed by W. Wolfer, U. of Wisconsin.

Based on the results of the topical report, a detailed research program (@ a general program plan) should be drafted. Necessary gaps in existing data and understanding should be filled by specific tests conducted within the context of the ADIP program.

Following the draft, the combined report should be circulated within DAFS and ADIP for comnent and review. Following revision, the final report should be issued and the research program plan implemented expeditiously.

1) Comment by A. S. Argon, M.I.T.

The current emphasis in fracture and fracture mechanics is on the accidental presence of cracks as pre-existing manufacturing defects in structures and is then based on considerations of how such cracks may grm under external stimuli such as the level of cyclic and mean stresses, environment, temperature, etc. This is normally a sound

strategy. On the other hand, it has been recognized by some, and soon by many others, that as manufacturing techniques and methods of non-destructive evaluation improve, structures will contain few if any such large cracks that obey the laboratory meusurcments designed for their growth by fracture mechanics techniques. Then the lives of parts will be governed by more inherent processes of crack formation, and large fractions of the useful life of a part will be spent in surface fragmentation and the growth of short cracks which obey only minimally the fracture mechanics laws. This more traditional and sound area of fracture research that had taken a hack seat upon the advent in 1964 of fracture mechanics based on large cracks is now belatedly returning again to prominence. There have been important advances made in the development and study of crack formation in cyclic deformation and in the presence of corrosion as well as in the development and growth of short cracks until they become large enough to grow by fracture mechanics. In fact, many structures are considered failed when they contain only short cracks. For example, large turbine generator shafts in thermal power stations are considered failed when they have cracks only as long as 0.25 to 0.375 inches - just when they become uorthy of study by fracture mechanics experts. There have been important recent developments in this urea both in Japan and in some quarters in the U.S.A. on the study of short cracks. An international conference was held last January in Asilomar, California, on this subject alone. Thus, in view of this, I feel it is essential that this area be included in the research plans for the fusion reactors where I believe these considerations will appear prominently, further accentuated by the special circumstances of the fusion environment.

2) Comment by I-Wei Chen, M.I.T.

The emphasis in this section is placed on the growth rate of surface cracks under near-threshold crack propagation conditions. It is,

however, not clear that design of the first wall materials can be based on the defect tolerant approach in which pre-existing cracks below a certain size are assumed to exist in adopting the method of fracture mechanics where the near-threshold fatigue correlation is introduced. As exemplified in our own experience in FCE analysis, any micron-size surface crack should remain essentially dormant wader rather substantial loading conditions when its growth is assessed using extrapolation of Paris law, which is thought to be too conservative for realistic applications by the report. The point here is that such short cracks cannot be handled competently by fracture mechanics modeling which is based on the continuous notion of deformation. Considering the thinness of first wall structures and the hostile environment, we believe that the design engineers will not tolerate any flaw in these structures which are of sufficient size for a fracture mechanics treatment. For this reason, the recommended attention to even the near-threshold fatigue crack growth may still be irrelevant. Instead, more emphasis should be placed on the crack initiation on the surface of thin sections under various environment and loading conditions.

3) Comment by O. K. Harling, M.I.T.

The enviroment of the first wall of CTRs is an unusually complex one, for which we must have a better understanding if we are to design and construct long-lived fusion power reactors. Important environmental parameters which can be expected to affect, mostly adversely, the first wall performance include:

a) high fast neutron fluxes (dpa's and transmutations),

b) Intense fluxes of heavy hydrogen species $\sim 10^{17}/cm^2$-sec, some with adequate energy (several hundred to several thousand eV) to penetrate into the near surface region of the first wall. Also Lower intensity fluxes of He will be present.

c) High energy alpha particle fluxes (MeV range) which can result from escape of alphas from confinement during the first orbit if there is no collision. Intensities of such alpha fluxes are system-dependent; however, fluxes of 10^{12} α/cm^2-sec are possible. These helium ions will drastically modify the first few microns by lattice damage and implantation. The implanted layer will have vast changes in mechanical properties and will develop eztremely high stresses. Microcracks, bubbles and enhanced trapping are some of the microphenomena which are known to occur.

d) The first wall, which is modified due to the radiation fluxes cited in a, b, and c, will be subjected, in some designs, to thermally-induced high cyclic stresses which combine with more constant coolant stresses. Reactor startups and shutdowns will introduce additional stresses and, in some cases, major material property changes, e.g., the DBTT in ferritic alloys. Plasma run-away and other off-normal operating conditions will occur and will, in some cases, produce very high stresses.

Because of the complexity of the situation, we have felt that properly designed scoping experiments would be useful. These would, hopefully, provide the material scientists and the design engineers with more understanding concerning the multiple simultaneous environmental parameters which operate at or near the first wall surface.

b. Provide Liaison With Other Task Groups on Stress Corrosion Cracking and Corrosion Fatigue (High Priority)

While chemically-induced failure modes such as static stress corrosion cracking (SCC) are generally important in design considerations, it is difficult to treat them generically. That is, SCC and similar failure modes are system-specific; the nature of the failure and the mechanisms vary widely among the many possible material/service environment combinations. Consequently, it is not a fundamental problem which can be addressed by OAFS in a general way.

However, it is important that DAFS be vigilant with respect to the treatment of environmental effects in the materials selection and the design considerations of planned fusion devices. For instance, for ETF it is important that IGSCC be a design consideration for austenitic stainless steels in contact with water; that H effects be considered for ferritic steel components; that an adequate data base exists in the literature to support these considerations, or that needed tests are performed if data are lacking; that potential effects of radiation fields or variable magnetic fields on electrochemical corrosion mechanisms are considered. It is not necessarily the task of DAFS to make these considerations, only to ensure that the considerations are made and to assist where necessary.

Consequently, it is recommended as a high priority that a person or group of persons within DAFS be identified as liaison between DAFS and the other components of the fusion program to ensure that chemically-induced fracture considerations are made when and where necessary and to ensure that assistance is provided when necessary in the accumulation of data and the understanding of problems of this nature. The best candidate to serve as the lead person in this liaison is R. Jones, BNWL, as he is both an active participant of DAFS and he is intimately familiar with chemically-assisted failure through his current research at Battelle, PNL.

c. Develop Models for Creep-Rupture (Medium Priority)

While creep rupture is a primary design consideration for high temperature devices, it is generally not a life limiting failure mode. Moreover, FED is not currently planned to operate at high temperature, so creep rupture is an even less likely failure mode for structural components in this facility. Consequently, while creep rupture is ultimately important as a consideration in the design of fusion machines, there is less urgency to address this problem in the near term than there is for other failure modes.

For the long term, ADIP will benefit from DAFS efforts to develop predictive methodology. While a considerable body of theory and data has been accumulated on the subject of creep rupture, a considerable number of uncertainties exist as well. Consequently, a simple fit to the data cannot be used as a predictive methodology. Instead, a methodology is required which is firmly rooted in physical understanding and calibrated against the best available data, i.e., a calibrated correlation model (CCM). The development and usefulness of CCMs for creep rupture thus require a reasonable understanding of the mechanism of creep rupture.

The important mechanisms leading to creep rupture have been identified as wedge-cracking at grain boundary junctions and cavitation along grain boundaries. While it is currently anticipated that the cavitation mechanism will be of more technological significance in fusion structural materials because of *the* high He inventories, in view of experience in the breeder reactor materials creep rupture program, wedge cracking cannot be discounted. It now appears that these mechanisms proceed in a very complex, interactive way which is sensitive *to* material microstructure and microchemistry both near and away from the grain boundaries and to test conditions such as stress state as well as stress and temperature. For instance, it is currently believed that while grain boundary cavities grow predominantly by stress-assisted vacancy diffusion, the actual rate of cavity growth (and hence creep rupture times) may be controlled by the rate at which localized stresses (built up by virtue of inhomogeneous vacancy production) are relieved; and this stress relaxation rate may in turn be controlled by 1) the rate of deformation around adjacent grain boundary precipitates, 2) the rate of sliding on grain boundaries adjacent to the cavitated boundary, or 3) the rate of matrix deformation in regions removed by several grain diameters from the cavitated grain boundary. As a consequence of these interactive characteristics, it does not seem feasible that physical understanding of the creep rupture mechanisms can be significantly advanced by separate effects studies. Instead, detailed microstructural evolution analyses on actual engineering alloys subjected to creep rupture conditions are required.

This is true of determining high energy neutron irradiation effects on creep rupture as well. Several irradiation effects may be anticipated to impact on creep rupture; namely, matrix hardening, irradiation creep, excess point defect populations, microchemical changes at the grain boundaries, and the creation of a significant He inventory. However, these effects will also be highly interactive, precluding separate effects studies of neutron irradiation creep rupture as well.

In view of the long-term need to develop calibrated correlation models for creep rupture in engineering alloys, and the need to advance our understanding of creep rupture processes in developing these CCMs, the following parallel path approach is recommended. This should have somewhat reduced priority fur the near term, but high priority for the long term.

1) The continued development of calibrated correlation models for creep rupture should be pursued. These CCMs should incorporate state-of-the-art understanding of creep rupture processes, and should be calibrated against FBR and ADIP data as it is produced.

   To support the CCM development, long-term plans should be formulated for detailed microstructural analysis of creep rupture specimens removed and placed into archives both prior to and after creep rupture. Consequently, ADIP specimens should be flagged in the near term for storage and ultimate use by DAFS in microstructural analyses. These should include unirradiated as well as irradiated specimens. If specimens cannot be sacrificed for early (pre-rupture) removal and storage, then DAFS-sponsored creep rupture tests should be planned to meet this need.

2) In addition to studying creep rupture with the current specimen design (pressurized tubes), an analysis should be made of alternate specimen designs to investigate stress-state effects on creep rupture. For instance, the current pressurized tube design may accommodate stress-state effects studies if near-end behavior

as opposed to tube mid-plane behavior is investigated; however, alternate designs such as square tubes, eccentric tubes, slit tubes, etc., may offer advantages or additional needed information over the current pressurized tube design. If such a need for additional designs is warranted, plans should be formulated to implement alternate designs in the ADIP program.

As a final, but lower priority consideration in the area of creep rupture, the concept of creep crack growth (as opposed to "break-away" cavitation or wedge cracking in the bulk of the material) as a failure mode should be kept in mind. There is a considerable literature developed to describe time-dependent failure by the growth of a preexisting flaw under high temperature, static load conditions. However, it is premature to devote considerable resources to investigate this failure mode for fusion systems as 1) there is already considerable fundamental research underway, supported by other programs, and 2) it is not clear that this is a relevant failure mode for thin-walled structural components for fusion systems. On the other hand, it is too early to discount this failure mechanism entirely.

3) Consequently, it is recommended that the evolving literature on creep crack growth be tracked over the near term. Moreover, based on the CCM development for creep rupture and the supporting data accumulated in the near term, a more enlightened decision on addressing the issue of creep crack growth should be made.

d. Evaluate the Relevance of Creep-Fatigue Interaction in Fusion Reactor First Wall Design (Medium Priority)

Despite a history of controversy, creep-fatigue interaction now appears to be a real time-dependent failure mechanism distinct from high temperature environment-fatigue interaction. That is, for a given set of fatigue test conditions in a high vacuum, as the frequency of loading is decreased the fatigue life may be decreased, particularly for certain

unbalanced load cycles. The problem may be aggravated by an agressive environment (such as air), but it persists in the absence of an environment at very low loading frequencies or long hold times.

It is unclear, however, whether a creep-fatigue interaction will be of importance in fusion systems design. Typically, for unirradiated alloys, creep-fatigue becomes an important consideration for component temperatures greater than 0.4 of the absolute melting temperature. However, the high He inventories anticipated for components close to the fusion reaction zone may lower this "transition" temperature considerably, particularly if the creep-fatigue interaction is governed by grain boundary cavity growth.

Again for near-term applications, creep-fatigue interaction is a research topic of low priority. However, for long term considerations, it is important to establish the relative importance of creep-fatigue interaction. Consequently, in the near term the following is recommended.

> A critical test, or series of tests, should be planned and executed to determine the relevance of creep-fatigue interaction as a time-dependent failure mechanism for fusion reactor components containing a high inventory of He in the constituent material. As a result of this test (or these tests) a recommendation for future work should be prepared.

D. RADIATION-INDUCED OR-ENHANCED EMBRITTLEMENT, Chairman: R. H. Jones, PNL

1. Introduction

The scope of the embrittlement working group was more restricted than the others, as it was limited to low temperature (25°C to ∿300°C) ductility and fracture of 316 SS, PCA and ferritic alloys. This subject differs from high-temperature fracture where time-dependent processes occur but does overlap with the "Effect of Flow on Fracture", since processes such as flow

localization can affect both low and high temperature ductility and fracture. In areas of obvious overlap, it was left to the group with the broader scope to list the recommended research.

The main causes of embrittlement which were considered by this group were:

- Radiation hardening
- Gaseous transmutation products, He and H
- Radiation or thermally activated segregation

2. Recommendations

a. Austenitic Stainless Steels

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

There is some evidence (Bloom and Wiffen) that helium decreases the ductility of annealed 316 SS without altering the yield strength at temperatures of 300°C to 350°C. This trend seems to extend to lower temperatures but there is no data to substantiate this at temperatures less than 300°C. The lack of a strengthening effect from helium at lower temperatures may be explained in part by the high hardening imparted by displacement damage masking any hardening from the helium.

Some of the concerns regarding the effect of helium on ductility at temperatures less than 350°C include:

- 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 the helium transported to the grain boundaries;
- is helium affecting the flow properties, i.e., flow localization, and, therefore, its ductility?

Answers to these and other questions about the role of helium in low temperature ductility is important in the development and selection of a material for FED applications.

2) Develop Property-Property Correlations (High Priority)

The need for property-property correlations is based primarily on the limited irradiation volume available for testing materials in fusion neutron environments such as RTNS II and FMIT. The availability of a materials testing module in a fusion reactor would reduce the need for these correlations, but the present fusion reactor development schedule indicates that fusion neutron sources such as RTNS II and FMIT will continue to play a significant role in materials testing.

Correlations between fracture and tensile properties are the highest priority because of the difficulty of irradiating fracture samples in FMIT; however, it is recognized that this is a very difficult correlation to make. Some progress has been made along these lines by G. R. Ddette, but more experimental data is needed to verify Ddette's approach, while other correlations should also be examined. Other property-property correlations of interest include hardness-yield strength, strength-ducility and yield strength-creep rate.

Equally important to the correlation model is the development of the appropriate samples to experimentally verify the correlations.

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

It is well recognized that grain boundary chemistry plays a significant role in both the low and high temperature fracture behavior of austenitic stainless steels. However, methods for evaluating the quantity of helium and impurities at the grain boundaries of austenitic stainless steels are not well developed. The problem is twofold:

a) Intergranular fracture must be accomplished at low temperatures in a surface chemistry probe in order to make meaningful grain boundary chemistry measurements.

b) Quantitative measurements of the elements present must be made. Slow bend or tensile tests of hydrogen-charged, notched samples of austenitic stainless steel have been successfully fractured in an Auger Electron spectrometer. It is recommended that such a capability be developed for evaluating irradiated fusion reactor materials. Semi-quantitative measurement of grain boundary impurities is within the present-day capabilities of surface chemistry probes, while similar capabilities do not exist for grain boundary helium. Use of a residual gas analyzer to measure the helium release on fracture may be adequate if very little helium remains adsorbed on the fracture surface.

4) <u>Write a Summary Report on Hydrogen Embrittlement of Austenitic Stainless Steels (Low Priority)</u>

Hydrogen embrittlement of austenitic steels has been observed both when a martensite phase is present in the steel and when no martensite phase can be detected. In the latter instance, grain boundary phosphorus was suspected as contributing to the embrittlement, while it was also suggested that hydrogen raises the Md of austenite, and therefore could contribute to the embrittlement by encouraging the austenite to martensite transformation with deformation. In any event, hydrogen embrittlement of austenitic stainless steels is a controversial topic but one which should be closely watched by the fusion materials community. It is, therefore, recommended that the literature on this subject be scrutinized (including the proceeding of the recent Third International Conference on the effect of Hydrogen on the Behavior of Materials) and a DAFS quarterly report be written summarizing these results.

b. Ferritic Steels

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

Embrittlement of ferritic steels by the segregation of trace impurities to grain boundaries is a well-established phenomena. In the steels of interest to fusion reactor designers, phosphorus and possibly sulfur segregation are of most concern. Both elements can cause embrittlement when as little as 0.02 monolayers is present at the grain boundaries of a high-strength steel. The complex interplay between the segregation kinetics of these elements and the quantity of chromium, molybdenum and manganese in the alloy adds to the uncertainty of embrittlement predictions for ferritic steels in fusion reactor blanket and first wall applications. Some questions which should be addressed include:

- Does irradiation significantly increase segregation of trace impurities?
- What is the relationship between strength increase due to irradiation- and segregation -induced embrittlement?
- Are the trace impurity limits for Cu, P, etc., established for pressure vessel steels adequate for fusion applications?

2) Develop Property-Property Correlations (High Priority)

The rationale for this effort is the same for austenitic and ferritic steels.

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

A ductile to brittle transition temperature shift of 108°C was observed by NRL for HT-9 irradiated to 10^{22} n/cm^2 at 420°C, while shifts of 45 to 90°C have been measured for 12 Cr steels irradiated to 6 x 10^{19} n/cm^2 at 300°C. Recent surveillance capsule results for pressure vessel steels suggest a saturation in ΔDBTT with fluence.

It is recommended that the surveillance pressure vessel data be evaluated for its relevance for fusion applications and that the change in DBTT with fluence be modeled or evaluated for fusion applications.

4) Evaluate the Potential for Hydrogen Embrittlement of Irradiated Materials (Low Priority)

It was felt that there was a need to "flag" the need to evaluate the hydrogen embrittlement or irradiated ferritic steels, as it is known the irradiation will increase the yield strength and that hydrogen embrittlement of ferritic steels requires less hydrogen with increasing yield strength. Also, segregation of trace impurities will enhance the sensitivity of a ferritic steel to hydrogen. Recent Sandia analysis suggests that hydrogen concentrations in ferritic steels will be less than 1 ppm, which is below the critical concentration for hydrogen embrittlement in unirradiated high-strength steels. However, two concerns still exist: a) what is the critical concentration for irradiated steels and b) what is the critical concentration to cause a sharp crack to propagate. Oriani and Josephic [1] and Gerberich [2] have shown that the threshold stress intensity depends on hydrogen pressure in the following way:

$$K_{TH} = C_1 - C_2 \text{ in } P_{H_2}.$$

Therefore, K_{TH} decreases with increasing hydrogen pressure. Changes in the strength of a steel by irradiation would presumably reduce C_1.

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