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- 2.5 TRANSMUTATION OF TUNGSTEN IN FFTF, HFIR AND STARFIRE -- F. A. Garner and
L. R. Greenwood (Pacific Northwest Laboratory) 71

Tungsten has been used in a variety of low activation ferritic alloys and also in copper composite alloys, both currently being irradiated in various fusion materials experiments. It has been proposed as an armor material also. However, in a manner that is strongly dependent on neutron spectra, tungsten transmutes strongly to rhenium and then osmium. This adds significant complexity to the interpretation of data developed in one spectral environment but intended for application to another environment.

- 2.6 SUMMARY OF IONIZING AND DISPLACIVE IRRADIATION FIELDS IN VARIOUS
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Calculations have been performed to estimate the ionizing and displacive irradiation fields that will occur in ceramics during irradiation in accelerators and fission and fusion reactors. A useful measure of the relative strength of ionizing vs. displacive radiation is the ratio of the absorbed ionizing dose to the displacement damage dose, which in the case of ion irradiation is equal to the ratio of the electronic stopping power to the nuclear stopping power. In ceramics such as Al₂O₃, this ratio is about 20 at a fusion reactor first wall, and has a typical value of about 100 in a fusion reactor blanket region and in mixed spectrum reactors such as HFIR. Particle accelerator sources typically have much higher ionizing to displacive radiation ratios, ranging from about 2000 for 1 MeV protons to >10,000 for 1 MeV electrons.

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2.8 ACTIVATION ANALYSES FOR DIFFERENT STRUCTURAL ALLOYS CONSIDERED FOR ITER--H. Attaya and D. Smith (Argonne National Laboratory) 85

Activation calculations have been made for the austenitic steel 316SS, the ferritic alloys HT-9, the titanium alloy Ti6Al4V, and the vanadium alloy V5Cr5Ti **in a liquid metal (Na) design** suggested recently for ITER. The calculations show that the vanadium alloy has the minimum short **and** long-term radioactivity and BHP. It **also has** the minimum decay heat all the time. The titanium alloy has less radioactivity than the austenitic and the ferritic alloys. However, the decay heat **of** this alloy could exceed that **of** the conventional alloys.

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4.1 MECHANICAL PROPERTIES ALONG INTERFACES OF BONDED STRUCTURES IN FUSION REACTORS -- M. H. Hassan and G. L. Kulcinski (University of Wisconsin) 95

Proper assessment of the mechanical properties along interfaces of bonded structures currently used in many fusion reactor designs is essential to compare the different fabrication techniques. **A** Mechanical Properties Microprobe (MPM) was used to measure hardness and Young's modulus along the interfaces of Be/Cu bonded structure. The MPM was able to distinguish different fabrication techniques by a direct measurement of the hardness, Young's **modulus, and** H/E^2 which reflects the ability of deformation of the interfacial region.

4.2 GREENS FUNCTION METHODOLOGY FOR FRACTURE MECHANICS OF **Sic-Sic COMPOSITE STRUCTURES -- A. El-Azab and N. M. Ghoniem (University of California, Los Angeles) 104**

A fundamental solution of plane elasticity in a finite domain is developed in this paper. A closed-form Green's function **for** **the** elastic field of an edge dislocation of arbitrary Burger's vector at an arbitrary point in an orthotropic **finite** elastic domain, that is free of **traction**, is presented. The method is **based** on the classical theory of potential fields, with an additional distribution of surface dislocations to satisfy the free traction boundary conditions. A solution is **first** developed for a dislocation in a semi-infinite half-plane. The resulting field is composed of two parts: a singular contribution from the original dislocation, and a regular component associated with the surface distribution. The Schwarz-Christoffel transformation **is** then utilized to map the field quantities to a finite, polygonal domain. A closed form solution containing Jacobi elliptic functions is developed for rectangular domains, and applications of the method to problems of fracture and plasticity are emphasized

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5.1 FACTORS WHICH CONTROL THE SWELLING OF Fe-Cr-Ni TERNARY AUSTENITIC ALLOYS --F. A. Garner and D. I. Edwards (Pacific Northwest Laboratory) 125

In agreement with limited earlier studies, a comprehensive irradiation experiment conducted in EBR-II demonstrates that while cold-working decreases swelling of Fe-15Cr-XNi (X=12-45 wt%) alloys at relatively low irradiation temperatures, it increases swelling at higher temperatures. Aging of cold-worked specimens to produce polygonized dislocation networks tends to further increase swelling, especially at higher nickel (>25%) levels. Swelling at lower nickel levels also appears to be sensitive to details of the annealing treatment.

5.2 THE INFLUENCE OF COLD WORK LEVEL ON SWELLING OF PURE COPPER IRRADIATED BY FAST NEUTRONS OR ELECTRONS -- F. A. Garner (Pacific Northwest Laboratory) and B. N. Singh (Risø National Laboratory) 127

Pure copper has been irradiated in a variety of starting conditions by either 1.0 MeV electrons or fast neutrons in FFTF-MOTA. Electron irradiation at 250 and 350°C produces a non-monotonic swelling behavior as a function of cold work level, increasing swelling at lower cold work levels and decreasing swelling at higher cold-work levels. In FFTF at 365 and 430°C, however, 10% cold-work reduces swelling initially, with little additional influence at higher cold-work levels. Swelling at 520 and 600°C is less than 1% at 35.9 and 13.6 dpa, respectively, with little effect of cold work level.

5.3 THE INFLUENCE OF DETAILS OF REACTOR HISTORY ON MICROSTRUCTURAL DEVELOPMENT DURING NEUTRON IRRADIATION -- F. A. Garner (Pacific Northwest Laboratory), N. Sekimura (Univ. of Tokyo), M. L. Grossbeck (Oak Ridge National Laboratory), A. M. Ermi (Westinghouse Hanford Company), J. W. Newkirk (U. of MO-Rolla), H. Watanabe (Kyushu University) and M. Kiritani (Nagoya University) 129

Microstructurally-oriented irradiation experiments are shown in this paper to be strongly dependent on details of reactor history that frequently are not brought to the experimenter's attention. In some cases, these details can dominate the experiment so as to produce very misleading results. To aid in the design and interpretation of microstructurally-oriented experiments, a number of studies are reviewed to highlight history effects and then guidelines are presented to minimize the impact of reactor history in new experiments.

5.4 NEUTRON-INDUCED SWELLING OF PURE NICKEL AND NICKEL BINARY ALLOYS -- F. A. Garner (Pacific Northwest Laboratory) 141

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Our knowledge of the processes involved in creating defects during cascade-producing irradiations is reviewed. Molecular dynamics simulations within the past few years have led to an understanding of the creation and survival of point defects in the critical first picoseconds of the cascade process, through the quenching of the thermal spike. The concept that "freely migrating defects" arise only from isolated Frenkel pairs produced in a cascade is critically discussed.

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6.1.1 EFFECTS OF HYDROGEN AND LOADING MODE ON THE FRACTURE TOUGHNESS OF A REDUCED ACTIVATION FERRITIC STAINLESS STEEL -- H. Li (Washington State University), R. H. Jones (Pacific Northwest Laboratory), J. P. Hirth (Washington State University), and D. S. Gelles (Pacific Northwest Laboratory) 165

The full spectrum of fracture toughness (J integrals), including pure mode I, different mixed mode I/III and pure mode III, will be examined for a ferritic/martensitic stainless steel with 0.1C-8Cr-2W-0.2V-0.04Ta-Fe (by wt%), designated as F-82H. The J integrals of pure mode I (J_{IC}) and mixed mode I/III (J_{mixed}) are determined with single specimen method using standard compact tension specimens and modified compact tension specimens, respectively. The pure mode III integral is measured with multiple specimen method using "triple-paillleg" specimens. Effects of hydrogen on the J integrals of pure mode I and mixed mode I/III are also going to be studied. 9 ppm H (about 500 appm) is pre-charged into specimens cathodically. The preliminary results showed that addition of mode III Stress (shear stress) to mode I loading had a significant negative effect on the fracture toughness of F-82H.

6.1.2 IRRADIATION CREEP AND SWELLING OF THE REGION HEATS OF HT9 AND 9Cr-1Mo TO 208 DPA AT -400°C -- F. A. Gamer (Pacific Northwest Laboratory), M. B. Toloczko (University of California at Santa Barbara) and C. R. Eiholzer (Westinghouse Hanford Company) 171

The irradiation creep behavior of the fusion heats of HT9 and 9Cr-1Mo at -400°C has been measured to exposures as large as 208 dpa. HT9 is somewhat nonlinear in its response to hoop stress level in the range 0-200 MPa, but 9Cr-1Mo exhibits only slightly greater than linear behavior with stress level. The strain data of both alloys appear to include some contributions from precipitate-related density changes. Swelling may have occurred in 9Cr-1Mo.

6.1.3 EFFECT OF VANADIUM AND TITANIUM ON MECHANICAL PROPERTIES OF LOW-CHROMIUM, REDUCED-ACTIVATION FERRITIC STEELS--R. L. Klueh and D. J. Alexander 174

Tensile and Charpy impact tests were made on three normalized-and-tempered 2 1/4Cr-2WV (0.1% C) steels with 0.1, 0.25, and 0.5% V (all concentrations are in weight percent). Increasing vanadium from 0.1 to 0.25% increased the yield stress up to twenty percent. A

higher ductile-brittle transition temperature (DBTT) accompanied the higher strength of the 0.25% V steel when both were tempered at 700°C. Tempering at 750°C gave similar DBTTs. Increasing vanadium from 0.25 to 0.5% caused a slight increase in strength with a large decrease in toughness. Thus a balance between strength and impact toughness is achieved with an intermediate vanadium concentration. Addition of 0.02% Ti to 2 1/4Cr-0.25V, 2 1/4Cr-2W, and 2 1/4Cr-2W-0.25V (0.1% C) steels caused a yield stress decrease of 10 to 30%, which was attributed to the effect of titanium on the MC precipitate distribution. The strength loss was accompanied by an increase in impact toughness, which may also have been affected by a decrease in prior austenite grain size. Furthermore, there was little difference in the DBTT of the Ti-modified steels tempered at 700 or 750°C. If it were possible to use a Ti-modified steel tempered at 700°C, this might offset the strength advantage of steels without titanium, which have to be tempered at the higher temperature.

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Experiments have shown that Type 316 stainless steel is susceptible to heat-affected-zone (HAZ) cracking upon cooling when welded using the gas tungsten arc (GTA) process under lateral constraint. The cracking has been hypothesized to be caused by stress-assisted helium bubble growth and rupture at grain boundaries. This study utilized an experimental welding setup, which enabled different compressive stresses to be applied to the plates during welding. Autogeneous GTA welds were produced in Type 316 stainless steel doped with 256 ppm helium. The application of a compressive stress, 55 MPa, during welding suppressed the previously observed catastrophic cracking. Detailed examinations conducted after welding showed a dramatic change in helium bubble morphology. Grain boundary bubble growth along directions parallel to the weld was suppressed. The results suggest that stress-modified welding techniques may be used to suppress or eliminate helium-induced cracking during joining of irradiated materials.

6.2.2	RELATIONSHIP BETWEEN SWELLING AND IRRADIATION CREEP IN COLD WORKED PCA STAINLESS STEEL TO 178 DPA AT ~400°C -- M. B. Toloczko (University of California at Santa Barbara) and F. A. Garner (Pacific Northwest Laboratory) ...	200
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At 178 dpa and ~400°C, the irradiation creep behavior of 20% cold-worked PCA has become dominated by the creep disappearance phenomenon. The total diametral deformation rate has reached the limiting value of 0.33%/dpa at the three highest stress levels. The stress-enhancement of swelling tends to camouflage the onset of creep disappearance, however.

6.2.3	DENSITY CHANGES OBSERVED IN PURE MOLYBDENUM AND Mo-41Re AFTER IRRADIATION IN FTT/MOTA -- F. A. Garner and L. R. Greenwood (Pacific Northwest Laboratory)	205
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Pure molybdenum and Mo-41wt% Re, in both the 20% cold-worked and aged and the annealed and aged conditions, were irradiated in FTT/MOTA to exposures as high as 111 dpa. Pure molybdenum appears to approach a saturation swelling level that is independent of the starting state. Cold-worked and aged molybdenum initially swells at a higher rate than that of solution-annealed and aged molybdenum and overshoots the saturation level at lower irradiation temperatures. This requires that part of the accumulated swelling be removed to approach saturation, probably by void shrinkage. The alloy Mo-41Re exhibits a more complex behavior with the annealed and aged condition initially swelling faster, but eventually the density change of both conditions begins to turn downward and tends toward densification. The role of solid transmutation to Tc, Re, and Os is thought to be very

important in the irradiation behavior of these two metals. Calculations of transmutant generation are provided for FFTF, HFIR and STARFIRE spectra.

- 6.2.4 IRRADIATION CREEP OF RUSSIAN FEDERATION PRESSURIZED TUBES IN MOTA-2B -- F. A. Garner (Pacific Northwest Laboratory), C. R. Eiholzer (Westinghouse Hanford Company), E. V. Demina and L. I. Ivanov (Baikov Institute) 211

Irradiation has been completed in MOTA-2B for creep tubes constructed from a candidate reduced activation austenitic alloy supplied by the Baikov Institute in Moscow. The total swains are strongly dependent on irradiation temperature in the range 425-600°C, but are not completely linear with stress level.

- 6.2.5 DENSITY MEASUREMENTS PERFORMED ON ISPRA SECOND GENERATION AMCR ALLOYS IRRADIATED IN MOTA-2A -- F. A. Garner (Pacific Northwest Laboratory), P. Schiller (Ispra Establishment), and H. Takahashi (Hokkaido University) 216

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- 6.2.6 THERMAL STABILITY OF MANGANESE-STABILIZED STAINLESS STEELS-- R. L. Klueh and E. A. Kenik (@& Ridge National Laboratory) 218

Previous work on a series of experimental high-manganese reduced-activation austenitic stainless steels demonstrated that they had improved tensile properties relative to type 316 stainless steel in both the annealed and 20% cold-worked conditions. Steels were tested with an Fe-20Mn-12Cr-0.25C (in weight percent) base composition, to which various combinations of Ti, W, V, P, and B were added. Tensile tests have now been completed on these steels after thermal aging at 600°C. Thermal stability varied with composition, but the alloys were as stable or more stable than type 316 stainless steel. The strength of the annealed steels increased slightly after aging to 5000 h, while a strength decrease occurred for the cold worked steel. In both conditions, a steel containing a combination of all the alloying elements was most stable and had the best strength after thermal aging 5000 h at 600°C. Despite having much higher strength than 316 stainless steel after aging, the ductility of the strongest experimental alloy was still as good as that of 316 stainless steel.

- 6.2.7 COMPARISON OF MICROSTRUCTURAL EVOLUTION IN REACTOR-IRRADIATED AUSTENITIC STAINLESS STEELS WITH AND WITHOUT SPECTRAL TAILORING - S. Jitsukawa, T. Sawai, K. Shiba, S. Hamada, and A. Hishinuma (Japan Atomic Energy Research Institute) 233

The effects of He/dpa ratio on swelling behavior were examined on three austenitic stainless steels. Materials were solution-annealed JPCA and two low carbon containing alloys (C and K) modified with titanium and niobium. These steels were neutron-irradiated in ORR and HFIR with and without spectrally tailoring, respectively. Achieved damage level was 7.4 dpa in ORR irradiation with average He/dpa of about 21 appm/dpa. In case of HFIR irradiation, they were 33 dpa and 76 appm/dpa, respectively. Alloy to alloy variation and temperature dependence of swelling behavior are far more distinctively detected in ORR irradiation than in HFIR irradiation, in spite of the lower damage level of ORR irradiation. In the case of ORR irradiation, JPCA exhibited small swelling values of <0.01 and 0.03% at 603 and 673 K, respectively, while a low carbon alloy K showed relatively larger swelling: 0.2% at 603 K and 0.6% at 673 K. Number densities of cavities in HFIR-irradiated alloys were larger than those observed in ORR by one to two orders. On the other hand, number densities and sizes of dislocation loops produced by ORR irradiation were two to five times as large as those by HFIR irradiation. These facts suggest that in ORR condition with closer He/dpa to

that of fusion, mutual annihilation rate of point defects was reduced and then bias driven cavity growth might be enhanced compared with HFIR condition.

- 6.2.8 MICROSTRUCTURES OF A WELDED JOINT USING AN IRRADIATED WRAPPER TUBE--S. Hamada, K. Watanabe, A. Hishinuma, I. Takahashi and T. Kikuchi (Japan Atomic Energy Research Institute) 241

The behavior of helium in welded joint fabricated using tungsten inert gas (TIG) welding process for a type 316 stainless steel wrapper tube irradiated in a fast reactor was investigated. The wrapper tube was irradiated to $(1.5 - 4.2) \times 10^{26} \text{ n/m}^2$ (helium level of 3 to 9 appm) at 395 - 410°C. All welded joints fractured in the heat-affected zone (HAZ). The microstructures of each portion of the base metal, the HAZ and the fusion zone in a welded joint were examined through a transmission electron microscope. Small helium bubbles were observed in number density of $2 \times 10^{20} \text{ m}^{-3}$ in the matrix and rarely found on the grain boundaries of the base metal. In the HAZ, small and large helium bubbles mixed and lined up along the grain boundaries. In particular, some of them elongated along the grain boundary. In the matrix of the fusion zone, delta-ferrite phases and unresolved carbides were scattered. Large cavities were attached to these precipitates and also occurred along grain boundaries. These results suggest that the failure in the HAZ of welded joints is attributed to the preferential growth and coalescence of helium bubbles in the grain boundaries of the HAZ caused by weld heat input and stress during welding

- 6.2.9 DOSE DEPENDENCE OF THE MICROSTRUCTURAL EVOLUTION IN NEUTRON-IRRADIATED AUSTENITIC STAINLESS STEEL -- S. J. Zinkle, P. J. Maziasz, and R. E Stoller (Oak Ridge National Laboratory) 251

Microstructural data on the evolution of the dislocation loop, cavity, and precipitate populations in neutron-irradiated austenitic stainless steels are reviewed in order to estimate the displacement damage levels needed to achieve the "steady state" condition. The microstructural data can be conveniently divided into two temperature regimes. In the low temperature regime (below about 300°C) the microstructure of austenitic stainless steels is dominated by "black spot" defect clusters and faulted interstitial dislocation loops. The dose needed to approach saturation of the loop and defect cluster densities is generally on the order of 1 displacement per atom (dpa) in this regime. In the high temperature regime (-300 to 700°C) cavities, precipitates, loops in excess of 10 dpa are generally required to approach a "steady state" microstructural conditions. Due to complex interactions between the various microstructural components that form during irradiation, a secondary transient regime is typically observed in temperatures. This slowly evolving secondary transient may extend to damage levels in excess of 50 dpa in typical 300-series stainless steels, and to >100 dpa in radiation-resistant developmental steels. The detailed evolution of any given microstructural component in the high-temperature regime is sensitive to slight variations in numerous experimental variables, including heat-to-heat composition changes and neutron spectrum.

- 6.2.10 FRACTURE TOUGHNESS OF IRRADIATED CANDIDATE MATERIALS FOR ITER FIRST WALL/BLANKET STRUCTURES: PRELIMINARY RESULTS -- D. J. Alexander, I. E. Pawel, M. L. Grossbeck, and A. F. Rowcliffe (Oak Ridge National Laboratory) 277

Candidate materials for first wall/blanket structures in ITER have been irradiated to damage levels of about 3 dpa at temperatures of either 60 or 250°C. Preliminary results have been obtained for several of these materials irradiated at 60°C. The results show that irradiation at this temperature reduces the fracture toughness of austenitic stainless steels, but the toughness remains quite high. The unloading compliance technique developed for the subsized disk compact specimens works quite well, particularly for materials with lower toughness. Specimens of materials with very high toughness deform excessively, and this results in experimental difficulties.

6.3 REFRACTORY METAL ALLOYS, 287

- 6.3.1 DENSITY CHANGES INDUCED BY NEUTRON IRRADIATION IN DYNAMICALLY COMPACTED TUNGSTEN AND PCA -- F. A. Garner (Pacific Northwest Laboratory) and J. Megusar (Massachusetts Institute of Technology) 289

Dynamically compacted tungsten with a starting density of 95.3% of the theoretical value densified 2 to 3% when irradiated in FFTF/MOTA-2A at three temperatures between 423 and 600°C and displacement levels corresponding to 32 and 36 dpa in stainless steel. Rapidly solidified and dynamically compacted PCA with high levels of titanium and carbon were also irradiated at these conditions. The density changes were small enough to determine that significant swelling had not occurred but, microscopy is necessary to determine whether void growth occurred in addition to precipitate-related strains.

- 6.3.2 DENSITY CHANGES OBSERVED IN Nb-1Zr AFTER IRRADIATION IN FFTF-MOTA -- F. A. Garner (Pacific Northwest Laboratory) 291

Nb-1Zr has been proposed for potential application to ITER. Whereas previous irradiation studies on Nb-1Zr were focused on the annealed condition, this study involved a comparative irradiation of both the annealed and aged, and the cold-worked and aged conditions. Based on measurements of density change, the cold-worked and aged condition appears to first undergo a phase-related dilation prior to the onset of void swelling, while the annealed condition densified prior to swelling and in some cases does not swell at all.

- 6.3.3 ASSESSMENT OF NIOBIUM-BASE ALLOYS FOR STRUCTURAL APPLICATIONS IN THE ITER DIVERTOR -- J. M. Purdy (Argonne National Laboratory) 294

The corrosion and embrittlement of pure Nb, Nb-1Zr, Nb-5Mo-1Zr, and Nb-5V-1.25Zr (alloy elements in wt.%) were evaluated in high-purity (HP) deoxygenated water at 300°C for up to 120 days. One heat of the Nb-5V-1.25Zr alloy ("O" lot) exhibited both a modest corrosion rate and good resistance to embrittlement relative to other Nb-base alloys. At present, Nb-5V-1.25Zr is the most promising Nb-base alloy on the basis of both corrosion and embrittlement characteristics in HP deoxygenated water at 300°C.

- 6.3.4 MICROSTRUCTURAL EVOLUTION INDUCED BY BORON TRANSMUTATION IN NEUTRON-IRRADIATED VANADIUM-BASE ALLOYS -- H. M. Chung (Argonne National Laboratory) 299

Microstructural evolution associated with transmutation of ^{10}B to helium and lithium has been characterized to provide a better understanding of the boron-doping technique, frequently used to simulate the effect of helium generation under fusion reactor conditions. Transmission electron microscopy (TEM) was used to examine specimens of V-20Ti alloy after irradiation at 600°C to $\approx 44\text{--}80$ dpa in the Fast Flux Test Facility (FFTF). In the earlier stage of irradiation to low fluence, concentric shells of He-damage and Li-damage zones are produced around a V_3B_2 precipitate or a ^{10}B -rich cluster. On further irradiation, helium atoms diffuse away from the damage shell either to be dissolved in the matrix or to form microcavities, leaving a shell rich in Li, defect clusters, and dislocations. Oxygen atoms in solid solution migrate toward the Li-rich shells, and $\gamma\text{-LiV}_2\text{O}_5$ shells precipitate subsequently. In view of this behavior, neither boron nor Li produced from the transmutation is likely to result in a detrimental weakening of grain boundaries.

- 6.3.5 STATUS OF THE DYNAMIC HELIUM CHARGING EXPERIMENT (DHCE)--H. Tsai, H. M. Chung, B. A. Loomis, D. L. Smith (Argonne National Laboratory), H. Matsui (Tohoku University), M. L. Hamilton, L. R. Greenwood, and R. Erni (Pacific Northwest Laboratory) . . . 306**

Irradiation of the seven DHCE capsules was completed in the Materials Open Test Assembly (MOTA)-2B at the end of Cycle 12B in the Fast Flux Test Facility (FFTF). The accrued exposure was 203.3 effective full-power days (EFPDs), vis-a-vis the target exposure of 300 EFPDs. Peak damage in the samples was ≈ 29 displacement per atom (dpa). All seven capsules have been discharged from the FFTF and are being shipped to Argonne National Laboratory (ANL), where the samples will be retrieved from the capsules and distributed to the experimenters, including Monbusho of Japan, for examination and testing. A substantial effort is underway at ANL to retrieve the samples from the highly tritiated capsules.

- 6.3.6 THERMAL CREEP BEHAVIOR OF V-5CR-5TI AND V-10CR-5TI ALLOYS -- H. M. Chung, B. A. Loomis, L. J. Nowicki, and D. L. Smith (Argonne National Laboratory) . . . 309**

The thermal creep rates and stress-rupture life of V-5Cr-5Ti and V-10Cr-5Ti alloys were determined at 600°C and the impurity composition and microstructural characteristics of creep-tested specimens were analyzed and correlated with the measured creep properties. The results of these tests show that V-5Cr-5Ti, which contains impurity compositions typical of a commercial vanadium-base alloy, exhibits creep strength substantially superior to that of V-20Ti, HT-9, or Type 316 stainless steel. The V-10Cr-5Ti alloy exhibits creep strength somewhat higher than that of V-5Cr-5Ti.

- 6.3.7 DUCTILE-BRITTLE TRANSITION TEMPERATURES OF UNIRRADIATED VANADIUM ALLOYS, BASED ON CHARPY-IMPACT TESTING--B. A. Loomis, L. J. Nowicki, J. Gazda, and D. L. Smith (Argonne National Laboratory) . . . 318**

Ductile-brittle transition temperatures (DBTTs) were determined by Charpy-impact tests for dehydrogenated and hydrogenated V-3Ti, V-5Cr-3Ti, and V-5Cr-5Ti alloys. These DBTT data complement the data previously obtained by Loomis et al. on Charpy-impact testing of unalloyed V, V-1Ti, V-3Ti-1Si, V-5Ti, V-10Ti, V-18Ti, V-4Cr-4Ti, V-8Cr-6Ti, V-9Cr-5Ti, V-10Cr-9Ti, V-14Cr-5Ti, V-15Cr-5Ti, V-7Cr-15Ti, and Vanstar-7 alloys. The results show that V alloys with Ti additions (0-18 wt.%) have a minimum DBTT ($\approx 250^\circ\text{C}$) in an alloy containing 3-5 wt.% Ti, that addition of 4 to 15 wt.% Cr to V-(4-6)Ti alloy results in a substantial increase (25-215°C) of the DBTT, and that 0.5 and 1.0wt.% Si additions to V-3Ti alloy result in a significant increase ($\approx 100^\circ\text{C}$) in DBTT. In addition, the results show that the presence of 400-1200 appm H in unalloyed V and V-base alloys causes a significant increase (≈ 400 ppin O, ≈ 200 ppin C, and ≈ 900 ppin Si for use as structural material in a fusion reactor.

- 6.4 COPPER ALLOYS . . . 327**

- 6.4.1 STATUS OF FATIGUE STUDIES ON IRRADIATED COPPER ALLOYS -- F. A. Garner and M. L. Hamilton (Pacific Northwest Laboratory), J. F. Stubbins and A. Singhal (University of Illinois), and B. N. Singh (Risø National Laboratory) . . . 329**

Irradiation continues in the EBR-II and III<-3 reactors of pure copper and GlidCop CuAl25 in the form of subsize tensile fatigue specimens. The 1st phase of the EBR-II irradiation sequence has been completed. A size effects experiment conducted on unirradiated CuAl25 fatigue specimens is nearing completion. Early results on the fatigue behavior of subsize specimens are presented in this report.

6.4.2 The Response of Dispersion-Strengthened Copper Alloys to High Fluence Neutron Irradiation at 415°C -- D. J. Edwards, J. W. Newkirk (Univ. of MO-Rolla), F. A. Garner, M. L. Hamilton (Pacific Northwest Laboratory), A. Nadkarni, and P. Samal (SCM Metal Products) 331

Various oxide-dispersion-strengthened copper alloys have been irradiated to 150 dpa at 415°C in the Fast Flux Test Facility (FFTF). The Al₂O₃-strengthened GlidCop™ alloys, followed closely by a HfO₂-strengthened alloy, displayed the best swelling resistance, electrical conductivity, and tensile properties. The conductivity of the HfO₂-strengthened alloy reached a plateau at the higher levels of irradiation, instead of exhibiting the steady decrease in conductivity observed in the other alloys. A high initial oxygen content resulted in significantly higher swelling for a series of castable oxide-dispersion-strengthened alloys, while a Cr₂O₃-strengthened alloy showed poor resistance to radiation.

6.4.3 NEUTRON-INDUCED SWELLING OBSERVED IN COPPER ALLOYS IRRADIATED IN MOTA's 2A AND 2B -- E. A. Garner (Pacific Northwest Laboratory), D. J. Edwards (University of Missouri-Rolla), B. N. Singh (Risø National Laboratory), and H. Watanabe (Kyushu University) 34s

Density measurements have been completed on copper alloys irradiated in MOTA 2A and MOTA 2B at (375°C, 12.7 dpa and 21.2 dpa) and (423°C, 48.0 and 95.4 dpa). While most of the density changes observed are consistent with those of earlier studies, there were several surprises. Addition of 5% Ni appears to accelerate the swelling rate initially at 423°C, but depresses swelling at 375°C. The suppressing action of cold work on swelling of Cu-5Ni is relatively small, and Cu-5Mn resists swelling very strongly in both the annealed and cold-worked conditions.

6.4.4 IRRADIATION OF COPPER ALLOYS IN THE SM-3 REACTOR--S. J. Zinkle (Oak Ridge National Laboratory), F. A. Garner (PNL), V. R. Barabash (D.V. Efremov), S. A. Fahritsiev (D. V. Efremov) and A. S. Pokrovsky (SRIAR) 347

A total of 74 alloys of varying composition and thermomechanical condition have been prepared for a joint US-Russia irradiation experiment in the SM-3 reactor in Dimitrograd, Russia. The alloys will be irradiated in the form of TEM disks and sheet tensile specimens at temperatures of about 120, 250, and 340°C for one 45-day cycle in the core and Channel 2 irradiation positions. This will produce damage levels of about 7 and 1 dpa, respectively. Cadmium shielding will be used in the Channel 2 position to shield the thermal neutrons and thereby reduce the amount of solid transmutation products in copper.

6.4.5 EFFECT OF ION IRRADIATION ON THE STRUCTURAL STABILITY OF DISPERSION-STRENGTHENED COPPER ALLOYS -- S. J. Zinkle (Oak Ridge National Laboratory), E. V. Nesterova, and V. V. Rybin (Central Research Institute for Structural Materials), and V. R. Barabash and A. V. Naberenskov (D. V. Efremov Scientific Research Institute of Electrophysical Apparatus) 352

Transmission electron microscopy was used to compare the microstructure and particle distributions of two commercial oxide dispersion-strengthened copper alloys, GlidCop A125 and MAGT 0.2. Measurements were made on specimens in their as-wrought condition, after thermal annealing for 1 h at 900°C, and after 3 MeV Ar⁺ ion irradiation at 180 and 350°C to damage levels of 20 to 30 displacements per atom (DPA). All of the annealed and ion-irradiated specimens were found to be resistant to recrystallization. In addition, void formation was not observed in any of the irradiated specimens. The GlidCop oxide particle geometry was transformed from triangular platelets to circular disks by the ion irradiation. The MAGT particle geometry consisted of circular disks and spheres before and after irradiation. The oxide particle edge length in the unirradiated GlidCop alloy was about 10 nm, whereas the mean particle diameter in both the MAGT and GlidCop alloys was about 6 nm.

- 6.4.2 The Response of Dispersion-Strengthened Copper Alloys to High Fluence Neutron Irradiation at 415°C -- D. J. Edwards, J. W. Newkirk (Univ. of MO-Rolla), F. A. Garner, M. L. Hamilton (Pacific Northwest Laboratory), A. Nadkarni, and P. Samal (SCM Metal Products). 331

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- 6.4.3 NEUTRON-INDUCED SWELLING OBSERVED IN COPPER ALLOYS IRRADIATED IN MOTA's 2A AND 2B -- F. A. Garner (Pacific Northwest Laboratory), D. J. Edwards (University of Missouri-Rolla), B. N. Singh (Risø National Laboratory), and H. Watanabe (Kyushu University) 345

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- 6.4.4 IRRADIATION OF COPPER ALLOYS IN THE SM-3 REACTOR--S. J. Zinkle (Oak Ridge National Laboratory), F. A. Garner (PNL), V. R. Barabash (D.V. Efremov), S. A. Fabritsiev (D. V. Efremov) and A. S. Pokrovsky (SRIAR) 347

A total of 74 alloys of varying composition and thermomechanical condition have been prepared for a joint ITS-Russia irradiation experiment in the SM-3 reactor in Dimitrovgrad, Russia. The alloys will be irradiated in the form of TEM disks and sheet tensile specimens at temperatures of about 120, 250, and 340°C for one 45-day cycle in the core and Channel 2 irradiation positions. This will produce damage levels of about 7 and 1 dpa, respectively. Cadmium shielding will be used in the Channel 2 position to shield the thermal neutrons and thereby reduce the amount of solid transmutation products in copper.

- 6.4.5 EFFECT OF ION IRRADIATION ON THE STRUCTURAL STABILITY OF DISPERSION-STRENGTHENED COPPER ALLOYS -- S. J. Zinkle (Oak Ridge National Laboratory), E. V. Nesterova, and V. V. Rybin (Central Research Institute for Structural Materials), and V. R. Barabash and A. V. Nabernekov (D. V. Efremov Scientific Research Institute of Electrophysical Apparatus) 352

Transmission electron microscopy was used to compare the microstructure and particle distributions of two commercial oxide dispersion-strengthened copper alloys. GlidCop A125 and MAGT 0.2. Measurements were made on specimens in their as-wrought condition, after thermal annealing for 1 h at 900°C, and after 3 MeV Ar⁺ ion irradiation at 180 and 350°C to damage levels of 20 to 30 displacements per atom (DPA). All of the annealed and ion-irradiated specimens were found to be resistant to recrystallization. In addition, void formation was not observed in any of the irradiated specimens. The GlidCop oxide particle geometry was transformed from triangular platelets to circular disks by the ion irradiation. The MAGT particle geometry consisted of circular disks and spheres before and after irradiation. The oxide particle edge length in the unirradiated GlidCop alloy was about 10

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6.5.1 MASS TRANSFER IN LITHIUM/STAINLESS STEEL TEST LOOP--P. R. Luebbers and O. K. Chopra (Argonne National Laboratory) 365

The plugged pipe removed from the cold-trap purification loop of the forced-circulation lithium system was examined to study mass transfer/deposition behavior and to establish the cause of plugging. Several intermetallic compounds were identified in residue collected from the plugged cold-trap pipe. Plugging was caused by deposition of calcium/zinc/nickel crystals in the pipe sections within the pump coil and flow-meter magnet. Addition of calcium as a getter to reduce the concentration of nitrogen in the lithium leads to formation of (Ca,Zn)Ni₅ crystals and subsequent plugging of the cold-trap loop. Deposits of manganese/iron/nickel globules and manganese/zinc/nickel dendrites, as well as Li₉CrN₅ and possibly Ca₃N₂, were also identified in the residue. These phases may have reduced flow through the cold-trap loop but were not abundant enough to plug the loop. The results indicate that the use of a dissolved getter, such as calcium, to reduce nitrogen content in an austenitic stainless steel loop may not be effective. Elements in the lithium from structural alloys (e.g., due to corrosion) and those added (e.g., calcium) to reduce the concentration of nonmetallic impurities (e.g., nitrogen) play an important role in the mass transfer/deposition behavior in circulating lithium systems.

6.5.2 COMPATIBILITY OF CANDIDATE STRUCTURAL MATERIALS WITH STATIC GALLIUM--P. R. Luebbers, W. F. Michaud, and O. K. Chopra (Argonne National Laboratory) . . 370

Scoping tests were conducted on compatibility of gallium with candidate structural materials. e.g., Type 316 SS, Inconel 625, and Nb-5 Mo-1 Zr alloy, as well as Armco iron, Nickel 270, and pure chromium. Type 316 stainless steel is least resistant and Nb-5 Mo-1 Zr alloy is most resistant to corrosion in static gallium. At 400°C, corrosion rates are ≈4.0, 0.5, and 0.03 mm/y for Type 316 SS, Inconel 625, and Nb-5 Mo-1 Zr alloy, respectively. The pure metals react rapidly with gallium. In contrast to findings in earlier studies, pure iron shows greater corrosion than does nickel. The corrosion rates at 400°C are ≥90 and 17 mm/y, respectively, for Armco iron and Nickel 270. The results indicate that at temperatures up to 400°C, corrosion occurs primarily by dissolution accompanied by formation of metal/gallium intermetallic compounds.

6.5.3 AQUEOUS STRESS CORROSION OF CANDIDATE AUSTENITIC STEELS FOR ITER STRUCTURAL APPLICATIONS--W. K. Soppet, D. M. French, and T. F. Kassner (Argonne National Laboratory) 380

Susceptibility of crevice-weld joint specimens of Types 316L and 316NG stainless steel (SS) to SCC was investigated in slow-strain-rate-tensile (SSRT) tests in water that simulates important parameters anticipated in first-wall/blanket systems. The SSRT tests were performed in oxygenated water containing 0.06-10 ppm chloride at temperatures of 95 to 225°C to establish the effects of water purity and temperature on SCC resistance. These steels, including weldments, exhibit good resistance to SCC under crevice conditions at temperatures of <150°C in water containing ≤0.1 ppm chloride. It appears that Type 316NG SS is somewhat more resistant to SCC than Type 316L SS at temperatures >150°C in oxygenated water containing 0.1-10 ppm chloride. Most specimens fractured in the base metal, and several others fractured in the heat-affected zone (HAZ) of the weld, but none failed in the weld metal.

6.5.4 FORMATION OF ELECTRICALLY INSULATING COATINGS ON ALUMINIDED VANADIUM-BASE ALLOYS IN LIQUID LITHIUM--J.-H. Park and G. Dragel (Argonne National Laboratory) 389

Aluminide coatings were produced on vanadium and vanadium-base alloys by exposure of the materials to liquid lithium that contained 3-5 at.% dissolved aluminum in sealed capsules at temperatures between 775 and 880°C. Reaction of the aluminide layer with dissolved nitrogen in liquid lithium provides a means of developing an in-situ electrical insulator coating on the surface of the alloys. The electrical resistivity of AlN coatings on aluminided V and V-20 wt.% Ti was determined in-situ.

6.5.5 CORROSION FATIGUE OF CANDIDATE AUSTENITIC STEELS FOR ITER STRUCTURAL APPLICATIONS--W. E. Ruther and T.F. Kassner (Argonne National Laboratory) 395

Crack-growth-rate (CGR) tests were performed on 1-in.-thick (1T) compact-tension (CT) specimens of Types 316NG and 316L stainless steel (SS) in oxygenated water containing 0-5 ppm Cl⁻ at 150, 185, and 225°C. The results obtained under cyclic loading conditions at stress intensity factors of ≈27 to 39 MPa·m^{1/2} indicate that environmental enhancement of the rates increases with Cl⁻ concentrations >0.1 ppm at 150°C in comparison with calculated rates in air under the specific loading conditions. In contrast, at the higher temperatures the CGRs were not affected by Cl⁻ in oxygenated water but were greater than the predicted rates in air by one order of magnitude.

6.5.6 DEVELOPMENT OF ALUMINIDE COATINGS ON VANADIUM-BASE ALLOYS IN LIQUID LITHIUM--J.-H. Park and G. Dragel (Argonne National Laboratory) 405

Aluminide coatings were produced on vanadium and vanadium-base alloys by exposure of the materials to liquid lithium that contained 3-5 at.% dissolved aluminum in sealed V and V-20 wt.% Ti capsules at temperatures between 775 and 880°C. After each test, the capsules were opened and the samples were examined by optical microscopy and scanning electron microscopy (SEM), and analyzed by electron-energy-dispersive spectroscopy (EDS) and X-ray diffraction. Hardness of the coating layers and bulk alloys was determined by microindentation techniques. The nature of the coatings, i.e., surface coverage, thickness, and composition, varied with exposure time and temperature, solute concentration in lithium, and alloy composition. Solute elements that yielded adherent coatings on various substrates can provide a means of developing in-situ electrical insulator coatings by reaction of the reactive layers with dissolved nitrogen in liquid lithium.

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No contributions

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8.1 CERAMICS RADIATION EFFECTS ISSUES FOR ITER -- S. J. Zinkle (Oak Ridge National Laboratory) 415

The key radiation effects issues associated with the successful operation of ceramic materials in components of the planned International Thermonuclear Experimental Reactor (ITER) are discussed. Radiation-induced volume changes and degradation of the mechanical properties should not be a serious issue for the fluences planned for ITER. On the other hand, radiation-induced electrical degradation effects may severely limit allowable the exposure of

ceramic insulators. Degradation of the loss tangent and thermal conductivity may also restrict the location of some components such as ICRH feedthrough insulators to positions far away from the first wall. In-situ measurements suggest that the degradation of physical properties in ceramics during irradiation is greater than that measured in postirradiation tests. Additional in-situ data during neutron irradiation are needed before engineering designs for ITER can be finalized.

8.2 EFFECT OF IRRADIATION SPECTRUM ON THE MICROSTRUCTURAL EVOLUTION IN OXIDE CERAMICS --S.J. Zinkle (Oak Ridge National Laboratory) 426

Cross section transmission electron microscopy was utilized to examine the radiation-induced microstructural changes in oxide ceramics after irradiation with a wide variety of ion beams. The microstructure showed a strong dependence on ion mass and energy. The microstructural results have been correlated with the calculated depth-dependent partitioning between ionization and displacement damage. This correlation indicates that defect clusters do not form in MgAl₂O₄ if the ratio of energy deposited into electronic ionization to atomic displacements is greater than about 10. The corresponding ratio needed to suppress defect cluster formation in MgO and Al₂O₃ is 500 to 1000. Additional microstructural evidence obtained on the ion irradiated ceramic specimens suggests that the physical mechanism responsible for the lack of defect clusters in highly ionizing radiation environments is associated with ionization-enhanced diffusion (IED), which promotes annihilation of the point defects created by displacement damage during the irradiation. The most important parameter for IED is the ratio of ionizing to displacive radiation, since U_{is} is roughly proportional to U_{te} amount of ionization per dpa. However, the absolute magnitude of the ionizing radiation flux is also important.

8.3 ELECTRICAL CONDUCTIVITY OF CERAMIC INSULATORS DURING EXTENDED ION IRRADIATION WITH AN APPLIED ELECTRIC FIELD--S. J. Zinkle (Oak Ridge National Laboratory) and W. Kesternich (Forschungszentrum Juelich) 437

The initial results are presented from a cyclotron ion irradiation program investigating radiation-induced conductivity (RIC) and radiation-induced electrical degradation (RIED) of ceramic insulators. Polycrystalline specimens of Al₂O₃, MgAl₂O₄, AlN and Si₃N₄ were irradiated with either 28 MeV He⁺⁺ or 20 MeV H⁺ ions at temperatures between 150 and 600°C with an applied dc electric field of 100 to 500 V/mm. A large prompt increase in the electrical conductivity was observed in all of the specimens during irradiation. However, there was no evidence for permanent electrical degradation in any of the specimens for damage levels up to about 5 x 10⁻³ displacements per atom.

8.4 IRRADIATION OF MgAl₂O₄ SPINEL IN FFTF-MOTA -- F. A. Garner and G. W. Hollenberg (Pacific Northwest Laboratory), C. A. Black and R. C. Bradt (University of Nevada-Reno) 447

MgAl₂O₄ spinel specimens irradiated in FFTF-MOTA at temperatures between 385 and 750°C to fluences ranging from 2.2 to 24.0 x 10²² n cm⁻² (E>0.1 MeV) darken significantly, but do not develop any loss in weight or change in dimensions. Measurements of Knoop hardness and its dependence on crystalline orientation, neutron fluence and irradiation temperature are in progress. Measurements of elastic properties are also nearing completion.

8.5 HIGH-TEMPERATURE PROPERTIES OF SiC/SiC FOR FUSION APPLICATIONS -- R. H. Jones and C. H. Henager, Jr. (Pacific Northwest Laboratory) 451

Si/SiC composites exhibit novel mechanical properties relative to their monolithic counterparts. The crack velocity (da/dt) versus stress intensity (K) relationship for monolithic

ceramics can be described by a simple power law relationship where K_I/K_{II} was found to exhibit a multi-stage da/dt versus K relationship similar to that for stress corrosion of metals. A K independent stage II was followed by a strongly K dependent stage III, which paralleled the monolithic behavior. Experiments to determine the threshold K or stage I were not conducted; however, it is expected that they exist for these materials. There is also evidence that the fracture resistance of these materials is greater if cracks are produced by subcritical growth processes relative to machined notches. Oxygen was found to increase da/dt and decrease the K for the stage II to stage III transition while cyclic loads produced little damage at low K values but there was some evidence for increasing damage with increasing number of cycles and K .

- 8.6 MEASUREMENT OF DC ELECTRICAL CONDUCTIVITY OF ALUMINA DURING SPALLATION-NEUTRON IRRADIATION -- E. H. Farnum, F. W. Clinard, Jr., J. C. Kennedy, III, W. E. Sommer, and M. D. Dammeyer (Los Alamos National Laboratory) . . . 457

An irradiation experiment was carried out during the summer of 1992 at the Los Alamos Spallation Radiation Effects Facility (LASREF). *In situ* measurements of electrical conductivity in alumina, sapphire, and mineral-insulated electrical cables were made at 640°C, 590°C, and 400°C. Both DC and AC (100 Hz to 1 MHz) measurements were made to a fluence of approximately $3 \times 10^{23} \text{ n/m}^2$. Optical absorption from 300 nm to 800 nm was measured in pure silica- and OH-doped silica-core optical fibers during the irradiation. A large number of passive samples were included in the irradiation, some at the furnace temperatures and some at ambient temperature. This report describes preliminary analysis of the DC conductivity measurements. The AC measurements are analyzed in the companion report.

- 8.7 MEASUREMENT OF AC ELECTRICAL CONDUCTIVITY OF SINGLE CRYSTAL Al_2O_3 DURING SPALLATION-NEUTRON IRRADIATION -- I. C. Kennedy, III, E. H. Farnum, W. E. Sommer, and F. W. Clinard, Jr. (Los Alamos National Laboratory) 465

Samples of single crystal Al_2O_3 , commonly known as sapphire, and polycrystalline Al_2O_3 were irradiated with spallation neutrons at the Los Alamos Spallation Radiation Effects Facility (LASREF) under various temperature conditions and with a continuously applied alternating electric field. This paper describes the results of measurements on the sapphire samples. Neutron fluence and flux values are estimated values pending recovery and analysis of dosimetry packages. The conductivity increased approximately with the square root of the neutron flux at fluences less than $3 \times 10^{19} \text{ n/m}^2$ at fluxes less than $1 \times 10^{16} \text{ n/m}^2\text{-sec}$. Conductivity initially decreased at low fluences with minimums near fluences of $1 \times 10^{20} \text{ n/m}^2$. Incubation periods with a gradual increase in conductivity preceded the onset of an accelerated increase in conductivity beginning at fluences as low as 10^{21} n/m^2 . The increase in conductivity reached saturation levels as high as $2 \times 10^{-2} (\text{ohm}\cdot\text{m})^{-1}$ at fluences as low as $2 \times 10^{22} \text{ n/m}^2$. Frequency swept impedance measurements indicated a change in the electrical properties from capacitive to resistive behavior with increasing fluence.

- 8.8 IRRADIATION EFFECTS IN CERAMICS: TRANSITION FROM LOW TO HIGH DOSE BEHAVIOR -- F. W. Clinard, Jr., and E. H. Farnum (Los Alamos National Laboratory) 475

Ceramics subjected to irradiation show a wide variety of damage responses, depending on composition, nature of bonding, crystal structure, impurity levels, starting microstructure, number of phases, and type of bombarding particle. As doses reach high levels (a condition that varies in magnitude from one material to another) major changes in physical properties can occur, and atomic arrangements may even change from crystalline to disordered. However, some ceramics show marked resistance to damage, and some properties may improve. More work is needed to fully understand these phenomena, but it is currently

possible in many cases to predict at least qualitatively both microstructural damage response and observed property changes.

- 8.9 THE EFFECT OF RADIATION INDUCED ELECTRICAL CONDUCTIVITY (RIC) ON THE THERMAL CONDUCTIVITY OF SAPPHIRE AT 77K -- D. P. White (Oak Ridge National Laboratory) 480

Microwave heating of plasmas in fusion reactors requires the development of microwave windows through which the microwaves can pass without great losses. The degradation of the thermal conductivity of alumina in a radiation environment is an important consideration in reliability studies of these microwave windows. Several recent papers have addressed this question at higher temperatures and at low temperatures. The current paper extends the low temperature calculations to determine the effect of phonon-electron scattering on the thermal conductivity at 77 K due to RIC. These low temperature calculations are of interest because the successful application of high power (>1 MW) windows for electron cyclotron heating systems in fusion reactors will most likely require cryogenic cooling to take advantage of the low loss tangent and higher thermal conductivity of candidate window materials at these temperatures.

- 8.10 SURFACE PREPARATION EFFECTS IN NEAR SURFACE MODULUS MEASUREMENT FOR CVD SiC -- M. Osborne (Rensselaer Polytechnic Institute), L. L. Snead (Oak Ridge National Laboratory), and D. Steiner (Rensselaer Polytechnic Institute) 484

Surface preparation has an observable effect on the data obtained from the Nanoindenter for shallow (20 nm) indents on CVD SiC when polished with Syton™. This observed effect is significantly less for 1/2 micron diamond polished CVD SiC and for deep (160 nm) indents. These effects were manifested by the relative variations in the experimental modulus and hardness data. An analytical analysis of the anticipated variation in the modulus and hardness is performed and shown to correlate well with the observed trends. The observed variations appear to be the result of SiC material properties, as well as surface preparation, since the predicted variations are much smaller than the observed variations.

- 8.11 MICROMECHANICS OF FIBER PULL-OUT AND CRACK BRIDGING IN SCS-6 SiC-CVD SiC COMPOSITE SYSTEM AT HIGH-TEMPERATURE -- A. El-Azah and N. M. Ghoniem (University of California, Los Angeles) 495

A micro mechanical model is developed to study fiber pull-out and crack bridging in fiber-reinforced SiC-SiC composites with time dependent thermal creep. By analyzing the creep data for monolithic CVD SiC (matrix) and the SCS-6 SiC fibers in the temperature range 900-1250°C, it is found that the matrix creep rates can be ignored in comparison to those of fibers. Two important relationships are obtained: (1) a time dependent relation between the pull-out stress and the relative sliding distance between the fiber and matrix for the purpose of analyzing pull-out experiments, and (2) the relations between the bridging stress and the crack opening displacement to be used in studying the mechanics and stability of matrix crack bridged by fibers at high temperatures. The present analysis can be also applied to Nicalon-reinforced CVD SiC matrix system since the Nicalon fibers exhibit creep characteristics similar to those of the SCS-6 fibers.