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- 1.4 TENSILE PROPERTIES OF V-(4-5)Cr-(4-5)Ti Alloys – H. M. Chung, L. Nowicki, D. Busch, and D. L. Smith (Argonne National Laboratory). 17

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- 1.6 KINETICS OF RECOVERY AND RECRYSTALLIZATION OF THE LARGE HEAT OF V-4Cr-4Ti – A. N. Gubbi, A. F. Rowcliffe, W. S. Eatherly, and L. T. Gibson (Oak Ridge National Laboratory). 29

A series of slow cycle and rapid cycle anneals was carried out in the large heat of V-4Cr-4Ti alloy (heat 832665). Also, a differential scanning calorimetry (DSC) study was initiated on the samples of the same alloy. The recovery and recrystallization phenomena of V-4Cr-4Ti in slow cycle annealing were quite different from that observed in rapid cycle annealing. The large driving force for recrystallization due to rapid heating resulted in the first nuclei appearing after only 1 minute at 1000°C. There was a two-stage hardness reduction; the first stage involved recovery due to cell formation and annihilation of dislocations, and second stage was associated with the growth of recrystallization nuclei. This is consistent with results obtained from the DSC in which there was a broad exothermic peak from ~200 to 800°C due to recovery followed by a sharp exotherm associated with recrystallization. The activation energy for recrystallization for V-4Cr-4Ti, which was determined as 576 ± 75 kJ/mole is significantly higher than that for pure V, and is thought to be related to Ti and Cr in solid solution.

- 1.7 EFFECT OF Cr AND Ti CONTENTS ON THE RECOVERY, RECRYSTALLIZATION, AND MECHANICAL PROPERTIES OF VANADIUM ALLOYS – A. N. Gubbi, A. F. Rowcliffe, D. J. Alexander, M. L. Grossbeck, W. S. Eatherly, and L. T. Gibson (Oak Ridge National Laboratory). 37

A series of vacuum-anneals at temperatures from 900 to 100°C for 1 to 4 h was carried out on small heats of vanadium alloys with Cr and Ti contents ranging from 2 to 6wt. %. The alloys examined in this work were V-3Cr-3Ti, V-4Cr-4Ti-Si, V-5Cr-5Ti, V-6Cr-3Ti, and V-6Cr-6Ti. Optical microscopy, TEM, and microhardness testing were conducted. Variation in Cr and Ti over the range 3 to 6 wt. % had no discernible effect on recovery/recrystallization behavior. The hardness of both recovered and recrystallized structures increased with total (Cr + Ti) content. In order to study the effect of Cr and Ti content on mechanical properties, Charpy impact testing and tensile testing were carried out on small heats of compositional variants. The V-4Cr-4Ti-Si alloy, in a fully recrystallized

conditions, exhibited a high level of resistance to cleavage failure with a DBTT at $\sim 190^{\circ}\text{C}$. The alloys containing higher concentrations of Cr and Ti, in a fully recrystallized condition, exhibited a DBTT around -100°C , whereas the V-3Cr-3Ti alloy failed by pure ductile shear at liquid nitrogen temperature without any ductile-to-brittle transition. Tensile testing was conducted on SS-3 tensile specimens punched from 0.762-mm-thick plates of V-3Cr-3Ti and V-6Cr-6Ti. The tests were done in air at room temperature at strain rates ranging from 10^{-3} to $2 \times 10^{-1}/\text{s}$. For V-6Cr-6Ti, both the 0.2% yield stress (YS) and the ultimate tensile strength (UTS) were higher than those for V-3Cr-3Ti at all strain rates. Both YS and UTS showed a similar trend of incremental increase with strain rate for the two alloys. In the same token, both alloys exhibited an identical behavior of almost no change in uniform and total elongation up to a strain rate of $10^{-1}/\text{s}$ and a decrease with further increase in strain rate.

- 1.8 EFFECT OF SPECIMEN SIZE ON THE FRACTURE TOUGHNESS OF V-4Cr-4Ti –
R. J. Kurtz (Pacific Northwest National Laboratory), Huaxin Li (Associated Western
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J-R curves were generated using the single specimen unload-compliance technique on four specimens of V-4Cr-4Ti to determine the effect of specimen dimensions on the fracture behavior. Ductile crack initiation and growth was observed in the 6.35 mm thick specimens but not in the 12.70 mm thick specimens. The J-R curves determined from these tests were not valid per ASTM validity criteria so quantitative measures of the resistance to ductile crack initiation and growth were not obtained. These data suggest that standard fracture toughness tests performed with small-scale DCT specimens may also not be valid.

- 1.9 EFFECT OF OXYGEN AND OXIDATION ON TENSILE BEHAVIOR OF V-5Cr-5Ti –
K. Natesan and W. K. Soppet (Argonne National Laboratory). 50

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- 1.10 CHEMICAL AND MECHANICAL INTERACTIONS OF INTERSTITIALS WITH
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Oxidation studies of V-4Cr-4Ti were conducted in air and reduced oxygen partial pressures (10^{-4} , 10^{-5} , and 10^{-6} torr). Reaction rates were determined by weight change measurements and chemical analyses. Mechanical properties after the exposures were determined by room temperature tensile tests.

In air at 400 and 500°C , oxide films form on the surface. Initially, rates are high but decrease with time reaching similar values to those found in oxygen partial pressures at 10^{-4} , 10^{-5} , and 10^{-6} torr. At 400°C , oxygen pick-up followed a logarithmic function of time and was confined to regions near the surface. Little change in room temperature tensile properties was noted for oxygen increases up to 1500 ppm. Thermal cycling specimens from 400°C to room temperature up to 14 times had no apparent effect on oxidation rate or tensile properties. At 500°C , oxygen pick-up appeared to follow a parabolic relation with time. Rates were ~ 10 times those at 400°C and correspondingly larger oxygen increases

occurred when compared with the 400°C tests after similar time periods. This resulted in a significant decrease in total elongation after 240 h.

At reduced oxygen partial pressures, rates were measured for times ≤ 100 h. Data are relatively sparse but generally show a slightly higher initial rate before slowing down. At 400°C increases to ~ 200 ppm oxygen were found with no effect on room temperature elongation. At 500°C increases in oxygen of ~ 2400 ppm after $50 \text{ h}/10^{-5}$ torr resulted in a decrease of around 25% in room temperature elongation. By comparison, exposure to air at 500°C for 12 h caused nearly the same results.

- 1.11 SOLUBILITY OF HYDROGEN IN V-4Cr-4Ti AND LITHIUM – J.-H. Park, G. Dragel, R. A. Erck, and D. L. Smith (Argonne National Laboratory) and R. E. Buxbaum (Michigan State University). 59

The solubility of hydrogen in V-4Cr-4Ti and liquid lithium was determined at 400-675°C and a hydrogen pressure of 1.76×10^{-4} torr (2.35×10^{-2} Pa). Hydrogen concentration in both materials decreased as temperature increased, and the ratio of the hydrogen concentration in liquid lithium and V-4Cr-4Ti (hydrogen distribution ratio R) increased with temperature, e.g., R was ≈ 17 at 400°C and ≈ 80 at 700°C. Desorption of hydrogen from V-4Cr-4Ti is a thermally activated process and the activation energy of the desorption rate is 0.405 eV.

- 1.12 HYDROGEN UPTAKE IN VANADIUM FIRST WALL STRUCTURES – E. P. Simonen and R. H. Jones (Pacific Northwest National Laboratory). 63

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- 1.13 CORROSION OF V AND V-BASE ALLOYS IN HIGH-TEMPERATURE WATER – I. M. Purdy, P. T. Toben, and T. F. Kassner (Argonne National Laboratory). 68

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- 1.14 CaO INSULATOR AND Be INTERMETALLIC COATINGS ON V-BASE ALLOYS FOR LIQUID-LITHIUM FUSION BLANKET APPLICATIONS – J.-H. Park and T. F. Kassner (Argonne National Laboratory). 73

The objective of this study is to develop (a) stable CaO insulator coatings at the Liquid-Li/structural-material interface, with emphasis on electrically insulating coatings that prevent adverse MHD-generated currents from passing through the V-alloy wall, and (b) stable Be-V intermetallic coatings for first-wall components that face the plasma. Electrically insulating and corrosion-resistant coatings are required at the liquid-Li/structural interface in fusion first-wall/blanket applications. The electrical resistance of CaO coatings produced on oxygen-enriched surface layers of V-5%Cr-5%Ti by exposing the alloy to liquid Li that contained 0.5-85 wt.% dissolved Ca was measured as a function of time at temperatures between 250 and 600°C. Crack-free Be₂V intermetallic coatings were also produced by exposing V-alloys to liquid Li that contained Be as a solute. These techniques can be applied to various shapes (e.g., inside/outside of tubes, complex geometrical shapes) because the coatings are formed by liquid-phase reactions.

- 1.15 EFFECT OF DYNAMICALLY CHARGED HELIUM ON TENSILE PROPERTIES OF V-5Ti, V-4Cr-4Ti, AND V-3Ti-1Si – H. M. Chung, B. A. Loomis, L. Nowicki, and D. L. Smith (Argonne National Laboratory). 77

In the Dynamic Helium Charging Experiment (DHCE), helium was produced uniformly in the specimen at linear rates of ≈ 0.4 to 4.2 appm He/dpa by the decay of tritium during irradiation to 18-31 dpa at 424-600°C in the lithium-filled DHCE capsules in the Fast Flux Test Facility. This report presents results of postirradiation tests of tensile properties of V-5Ti, V-4Cr-4Ti, V-3Ti-1Si. The effect of helium on tensile strength and ductility was insignificant after irradiation and testing at $>420^\circ\text{C}$. Contrary to initial expectation, room-temperature ductility of DHCE specimens was higher than that of non-DHCE specimens, whereas strength was lower, indicating that different types of hardening centers are produced during DHCE and non-DHCE irradiation. In strong contrast to results of tritium-trick experiments, in which dense coalescence of helium bubbles is produced on grain boundaries in the absence of displacement damage, no intergranular fracture was observed in any tensile specimens irradiated in the DHCE.

- 1.16 DENSITY DECREASE IN VANADIUM-BASE ALLOYS IRRADIATED IN THE DYNAMIC HELIUM CHARGING EXPERIMENT – H. M. Chung, T. M. Galvin, and D. L. Smith (Argonne National Laboratory). 83

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- 1.17 MICROSTRUCTURAL EVOLUTION OF V-4Cr-4Ti DURING ION IRRADIATION AT 200°C – J. Gazda and M. Meshii (Northwestern University), and B. A. Loomis and H. M. Chung (Argonne National Laboratory). 88

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- 1.18 UNUSUAL RESPONSE OF THE BINARY V-2Si ALLOY TO NEUTRON IRRADIATION IN FFTF AT 430-600°C – S. Ohnuki, H. Konoshita (Hokkaido University), F. A. Garner (Pacific Northwest National Laboratory), and H. Takahashi (Hokkaido University). 92

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- 2.2 REVIEW OF DATA ON IRRADIATION CREEP OF MONOLITHIC SiC – F. A. Garner, G. E. Youngblood, and M. L. Hamilton (Pacific Northwest National Laboratory). 98

An effort is now underway to design an irradiation creep experiment involving SiC composites and SiC fibers. In order to successfully design such an experiment, it is necessary to review and assess the available data for monolithic SiC to establish the possible bounds of creep behavior for the composite. The data available show that monolithic SiC will indeed creep at a higher rate under irradiation compared to that of thermal creep, and surprisingly, it will do so in a temperature-dependent manner that is typical of metals.

- 2.3 MICROSTRUCTURAL EFFECTS OF NEUTRON IRRADIATION ON SiC-BASED FIBERS – G. E. Youngblood and R. H. Jones (Pacific Northwest National Laboratory) and A. Hasegawa (Tohoku University, Japan). 101

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- 2.4 IMPROVEMENT OF THE THERMAL CONDUCTIVITY OF SiC_f/SiC COMPOSITE – G. E. Youngblood (Pacific Northwest Laboratory) and W. Kowbel (MER Corporation, Tucson, AZ). 107

The methods, high temperature annealing and doping, were examined for improving the thermal conductivity of simulated CVI/ β -SiC matrix material. For instance, a two hour 1500°C anneal led to the increase of the room temperature (RT) thermal conductivity from 38 to 59 W/mK. Be-doping was even more effective in causing the thermal conductivity to increase with RT conductivity values up to 160 W/mK attained. To further optimize the thermal conductivity, hot-pressed SiC materials with carefully controlled amounts of Be- and B₄C-doping were investigated. Although a small improvement ($\approx 8\%$) was achieved with 2.0 wt % Be-doping, the effort to refine the amount of doping needed was largely unsuccessful. Apparently, hot-pressing SiC introduced numerous substructural stacking faults which effectively scattered phonons in the intermediate temperature range and nullified the benefits of doping. Nevertheless, Be- and B₄C-doping and/or thermal treatments appear to be promising strategies to achieve the goal of eventually improving the thermal conductivity of SiC_f/SiC composite.

- 2.5 THE MONOTONIC AND FATIGUE BEHAVIOR OF CFCCs – N. Miriyala, P. K. Liaw, and C. J. McHargue (University of Tennessee) and L. L. Snead (Oak Ridge National Laboratory). 115

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3.7 DYNAMIC FINITE ELEMENT MODELING OF THE EFFECTS OF SIZE ON THE UPPER SHELF ENERGY OF FERRITIC STEELS – S. E. Sidener, A. S. Kumar (University of Missouri), L. E. Schubert, M. L. Hamilton (Pacific Northwest National Laboratory), and S. T. Rosinski (Electric Power Research Institute). 147

Both the fusion and light water reactor program require the use of subsized specimens to obtain sufficient irradiation data on neutron-induced embrittlement of ferritic steels. While the development of fusion-relevant size effects correlations can proceed analytically, it is more cost-effective at this time to use data currently being obtained on embrittlement of pressure vessel steels to test and expand the correlations developed earlier using fusion-relevant steels.

Dynamic finite element modeling of the fracture behavior of fatigue-precracked Charpy specimens was performed to determine the effect of single variable changes in ligament size, width, span, and thickness on the upper shelf energy. A method based on tensile fracture-strain was used for modeling crack initiation and propagation. It was found that the upper shelf energy of precracked specimens (USE_p) is proportional to b^n , where b is ligament size and n varies from about 1.6 for subsized to 1.9 for full size specimens. The USE_p was found to be proportional to width according to $W^{2.5}$. The dependence on thickness was found to be linear for all cases studied. Some of the data from the FEM analysis were compared with experimental data and were found to be in reasonable agreement.

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Specimen fabrication is underway for an irradiation screening experiment planned to start in January 1996 in the SM-2 reactor in Dimitrovgrad, Russia. The purpose of the experiment is to evaluate the effects of neutron irradiation at ITER-relevant temperatures on the bond integrity and performance of Cu/SS and Be/Cu joints, as well as to further investigate the base metal properties of irradiated copper alloys. Specimens from each of the four ITER parties (U.S., EU, Japan, and RF) will be irradiated to a dose of ~0.2 dpa at two different temperatures, 150 and 300°C. The specimens will consist of Cu/SS and Be/Cu joints in several different geometries, as well as a large number of specimens from the base materials. Fracture toughness data on base metal and Cu/SS bonded specimens will be obtained from specimens supplied by the U.S. Due to a lack of material, the Be/Cu specimens supplied by the U.S. will only be irradiated as TEM disks.

4.2 TENSILE AND ELECTRICAL PROPERTIES OF COPPER ALLOYS IRRADIATED IN A FISSION REACTOR – S. A. Fabritsiev (D.V. Efremov Institute, St. Petersburg, Russia), A. S. Pokrovsky (Scientific Research Institute of Atomic Reactors, Dimitrovgrad, Russia), S. J. Zinkle and A. F. Rowcliffe (Oak Ridge National Laboratory), D. J. Edwards and F. A. Garner (Pacific Northwest National Laboratory), V. A. Sandakov (Scientific Research Institute of Atomic Reactors, Dimitrovgrad, Russia), B. N. Singh (Risø National Laboratory, Roskilde, Denmark) and V. R. Barabash (ITER Joint Central Team, Garching, Germany). 177

Postirradiation electrical resistivity and tensile measurements have been completed on pure copper and copper alloy sheet tensile specimens irradiated in the SM-2 reactor to doses of ~0.5 to 5 dpa and temperatures between ~80 and 400°C. Considerable radiation hardening and accompanying embrittlement was observed in all of the specimens at irradiation temperatures below 200°C. The radiation-induced electrical conductivity degradation consisted of two main components: solid transmutation effects and radiation damage (defect cluster and particle dissolution) effects. The radiation damage component was nearly

constant for the doses in this study, with a value of $\sim 1.2 \text{ n}\Omega\text{-m}$ for pure copper and $\sim 1.6 \text{ n}\Omega\text{-m}$ for dispersion strengthened copper irradiated at $\sim 100^\circ\text{C}$. The solid transmutation component was proportional to the thermal neutron fluence, and became larger than the radiation damage component for fluences larger than $\sim 5 \times 10^{24} \text{ n/m}^2$. The radiation hardening and electrical conductivity degradation decreased with increasing irradiation temperature, and became negligible for temperatures above $\sim 300^\circ\text{C}$.

- 4.3 FRACTURE TOUGHNESS AND FATIGUE CRACK GROWTH OF OXIDE-DISPERSION STRENGTHENED COPPER – D. J. Alexander and B. G. Gieseke (Oak Ridge National Laboratory). 189

The fracture toughness and fatigue crack growth behavior of copper dispersion strengthened with aluminum oxide (0.15 wt % Al) was examined. In the unirradiated condition, the fracture toughness was about 45 kJ/m^2 ($73 \text{ MPa}\sqrt{\text{m}}$) at room temperature, but decreased significantly to only 3 kJ/m^2 ($20 \text{ MPa}\sqrt{\text{m}}$), at 250°C . After irradiation at approximately 250°C to about 2.5 displacements per atom (dpa), the toughness at room temperature was about 19 kJ/m^2 ($48 \text{ MPa}\sqrt{\text{m}}$), and at 250°C the toughness was very low, about 1 kJ/m^2 ($12 \text{ mPa}\sqrt{\text{m}}$). The fatigue crack growth rate of the unirradiated material at room temperature is similar to other candidate structural alloys such as V-4Cr-4Ti and 316L stainless steel. The fracture properties of this material at higher temperatures and in controlled environments need further investigation, in both irradiated and unirradiated conditions.

- 4.4 FATIGUE BEHAVIOR OF COPPER AND SELECTED COPPER ALLOYS FOR HIGH HEAT FLUX APPLICATIONS – K. D. Leedy and J. F. Stubbins (University of Illinois), B. N. Singh (Risø National Laboratory), and F. A. Garner (Pacific Northwest National Laboratory). 195

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- 4.5 EVALUATION OF Nb-BASE ALLOYS FOR THE DIVERTOR STRUCTURE IN FUSION REACTORS – I. M. Purdy (Argonne National Laboratory) and J. A. Todd (Illinois Institute of Technology). 196

Niobium-base alloys are candidate materials for the divertor structure in fusion reactors. For this application, an alloy should resist aqueous corrosion, hydrogen embrittlement, and radiation damage and should have high thermal conductivity and low thermal expansion. Results of corrosion and embrittlement screening tests of several binary and ternary Nb alloys in high-temperature water indicated that Nb-1Zr, Nb-5Mo-1Zr, and Nb-5V-1Z4 (wt.%) showed sufficient promise for further investigation. These alloys, together with pure Nb and Zircaloy-4, have been exposed to high-purity water containing a low concentration of dissolved oxygen ($<12 \text{ ppb}$) at 170, 230, and 300°C for up to $\approx 3200 \text{ h}$. Weight-change data, microstructural observations, and qualitative mechanical-property evaluations reveal that Nb-5V-1Zr is the most promising alloy at higher temperatures. Below $\approx 200^\circ\text{C}$, the alloys exhibit similar corrosion behavior.

- 5.0 AUSTENITIC STAINLESS STEELS. 201

- 5.1 EFFECTS OF LOW TEMPERATURE NEUTRON IRRADIATION ON DEFORMATION BEHAVIOR OF AUSTENITIC STAINLESS STEELS – J. E. Pawel, A. F. Rowcliffe, D. J. Alexander, M. L. Grossbeck (Oak Ridge National Laboratory), and K. Shiba (Japan Atomic Energy Research Institute). 203

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- 5.2 FRACTURE TOUGHNESS OF IRRADIATED CANDIDATE MATERIALS FOR ITER FIRST WALL/BLANKET STRUCTURES: SUMMARY REPORT – D. J. Alexander, J. E. Pawel, M. L. Grossbeck, and A. F. Rowcliffe (Oak Ridge National Laboratory), and K. Shiba (Japan Atomic Energy Research Institute). 204
- Disk compact specimens of candidate materials for first wall/blanket structures in ITER have been irradiated to damage levels of about 3 dpa at nominal irradiation temperatures of either 90 or 250°C. These specimens have been tested over a temperature range from 20 to 250°C to determine J-integral values and tearing moduli. The results show that irradiation at these temperatures reduces the fracture toughness of austenitic stainless steels, but the toughness remains quite high. The toughness decreases as the temperature increases. Irradiation at 250°C is more damaging than at 90°C, causing larger decreases in the fracture toughness. The ferritic-martensitic steels HT-9 and F82H show significantly greater reductions in fracture toughness than the austenitic stainless steels.
- 5.3 MICROSTRUCTURAL OBSERVATION OF HFIR-IRRADIATED AUSTENITIC STAINLESS STEELS INCLUDING WELDS FROM JP-9-16 – T. Sawai, K. Shiba, A. Hishinuma (Japan Atomic Energy Research Institute (JAERI)). 217
- Austenitic stainless steels, including specimens taken from various electron beam (EB) welds, have been irradiated in HFIR Phase II capsules, JP9-16. Fifteen specimens irradiated at 300, 400, and 500°C up to 17 dpa are so far examined by a transmission electron microscope (TEM). In 300°C irradiation, cavities were smaller than 2 nm and different specimens showed little difference in cavity microstructure. At 400°C, cavity size was larger, but still very small (<8 nm). At 500°C, cavity size reached 30 nm in weld metal specimens of JPCA, while cold worked JPCA contained only small (<5 nm) cavities. Inhomogeneous microstructural evolution was clearly observed in weld-metal specimens irradiated at 500°C.
- 5.4 IRRADIATION CREEP IN AUSTENITIC AND FERRITIC STEELS IRRADIATED IN A TAILORED NEUTRON SPECTRUM TO INDUCE FUSION REACTOR LEVELS OF HELIUM – M. L. Grossbeck and L. T. Gibson (Oak Ridge National Laboratory), and S. Jitsukawa (Japan Atomic Energy Research Institute). 223
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- 6.0 INSULATING CERAMICS AND OPTICAL MATERIALS. 225
- 6.1 INFLUENCE OF IRRADIATION SPECTRUM AND IMPLANTED IONS ON THE AMORPHIZATION OF CERAMICS – S. J. Zinkle and L. L. Snead (Oak Ridge National Laboratory). 227
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- 6.2 ELECTRICAL INTEGRITY OF OXIDES IN A RADIATION FIELD – S.J. Zinkle (Oak Ridge National Laboratory) and C. Kinoshita (Kyushu Univ.). 229
- Extended abstract.
- 6.3 LOSS TANGENT MEASUREMENTS ON UNIRRADIATED ALUMINA – S. J. Zinkl and R. H. Goulding (Oak Ridge National Laboratory). 231
- Unirradiated room temperature loss tangent data for sapphire and several commercial grades of polycrystalline alumina are compiled for frequencies between 10^5 and 4×10^{11} Hz. Sapphire exhibits significantly lower values for the loss tangent at frequencies up to 10^{11} Hz. The loss tangents of 3 different grades of Wesgo alumina (AL300, AL995, AL998) and 2 different grades of Coors alumina (AD94, AD995) have typical values near $\sim 10^{-4}$ at a frequency of 10^8 Hz. On the other hand, the loss tangent of Vitox alumina exhibits a large loss peak ($\tan \delta \sim 5 \times 10^{-3}$) at this frequency.

- 6.4 INVESTIGATION OF THE FEASIBILITY OF IN-SITU DIELECTRIC PROPERTY MEASUREMENTS ON NEUTRON-IRRADIATED CERAMIC INSULATORS – R. H. Goulding and S. J. Zinkle (Oak Ridge National Laboratory). 236

Computer modeling and experimental benchtop tests have demonstrated that a capacitively loaded resonant coaxial cavity can produce accurate in-situ measurements of the loss tangent and dielectric constant of ceramic insulators at a frequency of ~80 MHz during fission reactor irradiation. The start of the reactor irradiations has been postponed indefinitely due to budgetary constraints.

- 6.5 CAPSULE FABRICATION FOR IN-SITU MEASUREMENT OF RADIATION INDUCED ELECTRICAL DEGRADATION (RIED) OF CERAMICS IN HFIR – W. S. Eatherly, D. W. Heatherly, M. T. Hurst, A. L. Qualls, D. G. Raby, R. G. Sitterson, L. L. Snead, K. R. Thoms, R. L. Wallace, D. P. White, and S. J. Zinkle (Oak Ridge National Laboratory), E. H. Farnum and K. Scarborough (Los Alamos National Laboratory), T. Sagawa (JAERI), K. Shiyyama (Kyushu University), M. Narui and T. Shikama (Tohoku University). 241

A collaborative DOE/Monbuscho series of irradiation experiments is being implemented to determine, in-situ, the effects of irradiation on the electrical resistivity of ceramic materials. The first experiment, TRIST-ER1, has been designed to irradiate 15 Al₂O₃ test specimens at 450°C in an RB* position of the High Flux Isotope Reactor (HFIR). Each test specimen is located in a sealed vanadium subcapsule with instrumentation provided to each subcapsule to measure temperature and resistance, and to place a biasing voltage across the specimen. Twelve of the specimens will be biased with 200 V/mm across the sample at all times, while three will not be biased, but can be if so desired during the irradiation. The experiment design, component fabrication, and subcapsule assembly have been completed. A three cycle irradiation, to a fast neutron (E>0.1 MeV) fluence of about 3 × 10²⁵ n/m² (~3 dpa in Al₂O₃), is expected to begin early in March 1996.

- 6.6 ISEC-3: RESULTS FROM THE THIRD IN-SITU ELECTRICAL CONDUCTIVITY TEST ON POLYCRYSTALLINE ALUMINA L. L. Snead, D. P. White,* W. S. Eatherly, and S. J. Zinkle (Oak Ridge National Laboratory). 249

An experimental investigation of radiation induced electrical degradation (RIED) has been performed at the High Flux Beam Reactor (HFBR) at Brookhaven National Laboratory. In this study (the third in a series of experiments at the HFBR) the effects of neutron irradiation on the electrical conductivity of Wesgo AL995 polycrystalline alumina has been investigated at approximately 450°C. The capsule design used in this study is very similar to a design used in the first two experiments in this series with some improvements made in the cable terminations. A guard ring configuration was used on the disk shaped sample. Triaxial mineral insulated cable was used as the data lead from the sputter deposited guard ring and central electrode of the sample, and coaxial mineral insulated cable was used as the sample power lead. No evidence for RIED was observed in this series of experiments to a dose level of ~1.8 dpa. The effect of neutron irradiation on the electrical properties of two mineral insulated (MgO) cables was also investigated.

- 6.7 SUMMARY OF THE IEA WORKSHOP ON RADIATION EFFECTS IN CERAMIC INSULATORS – S. J. Zinkle (Oak Ridge National Laboratory). 258

A brief summary is given of research on radiation effects in ceramic insulators for fusion energy applications performed during the last two years in Europe, Canada, Japan, the Russian Federation, the Ukraine and the United States. The IEA round-robin radiation-induced electrical degradation (RIED) experiment on Wesgo AL995 polycrystalline alumina has been completed by 5 research groups, with none of the groups observing clear indications of RIED.

6.8	OPTICAL PROPERTIES OF SILICA FIBERS AND LAYERED DIELECTRIC MIRRORS – D. W. Cooke, E. H. Farnum, F. W. Clinard, Jr., B. L. Bennett (Los Alamos National Laboratory) and A. M. Portis (UC-Berkeley).	262
	<p>Radioluminescence (RL) from virgin and neutron-irradiated (10^{23} n-m⁻²) silica fibers has been measured in the temperature interval 4 to 300 K. Unirradiated specimens exhibit a <i>decrease</i> in RL intensity with increasing temperature such that the intensity is extremely weak at room temperature. The luminescence is well described by a barrier-limited exciton mechanism. In contrast, the heavily-irradiated samples show an <i>increase</i> in RL with elevated temperatures such that the intensity at room temperature is about twice that measured at 4 K. Neutron irradiation presumably produces many luminescence centers that act as radiative sites for exciton decay. Absolute specular reflectance of a series of neutron-irradiated, layered dielectric mirrors was also measured. In addition to structural damage that has already been reported, we typically found approximately 10% reduction in the reflectance following irradiation. These results suggest that neither fibers nor dielectric mirrors are well suited for use near the high radiation area of the ITER plasma.</p>	
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8.1	THEORY AND MODELING OF RADIATION EFFECTS IN MATERIALS FOR FUSION ENERGY SYSTEMS – H. L. Heinisch (Pacific Northwest National Laboratory).	271
	<p>The U.S./Japan Workshop on Theory and Modeling of Radiation Effects in Materials for Fusion Energy Systems, under Phase III of the DOE/Monbuscho collaboration, convened on July 17-18, 1995, at Lawrence Livermore National Laboratory. A brief summary of the workshop is followed by the workshop program.</p>	
8.2	DISPLACEMENT RATE DEPENDENCE OF IRRADIATION CREEP AS PREDICTED BY THE PRODUCTION BIAS MODEL – C. H. Woo (Atomic Energy of Canada Limited) and F. A. Garner (Pacific Northwest National Laboratory).	274
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8.3	STOCHASTIC ANNEALING SIMULATION OF CASCADES IN METALS – H. L. Heinisch (Pacific Northwest National Laboratory).	275
	<p>The stochastic annealing simulation code ALSOME is used to investigate quantitatively the differential production of mobile vacancy and SIA defects as a function of temperature for isolated 25 KeV cascades in copper generated by MD simulations. The ALSOME code and cascade annealing simulations are described. The annealing simulations indicate that above Stage V, where the cascade vacancy clusters are unstable, nearly 80% of the post-quench vacancies escape the cascade volume, while about half of the post-quench SIAs remain in clusters. The results are sensitive to the relative fractions of SIAs that occur in small, highly mobile clusters and large stable clusters, respectively, which may be dependent on the cascade energy.</p>	
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Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). These experiments were irradiated at 85 MW for 238.5 EFPD. The maximum fast neutron fluence >0.1 MeV was about $2.1E + 22$ n/cm² for all of the experiments resulting in about 17.3 dpa in 316 stainless steel.

9.2 NEUTRON DOSIMETRY AND DAMAGE CALCULATIONS FOR THE JP-17, 18, AND 19 EXPERIMENTS IN HFIR – L. R. Greenwood (Pacific Northwest National Laboratory) and C. A. Baldwin. 286

Neutron fluence measurements and radiation damage calculations are reported for the joint U.S./Japanese experiments (JP-17, 18, and 19 in the target of the High Flux Isotope Reactor (HFIR) at Oak Ridge National (ORNL). These experiments were irradiated at 85 MW for two cycles resulting in 43.55 EFPD for JP-17 and 42.06 EFPD for JP-18 and 19. The maximum fast neutron fluence >0.1 MeV was about $3.7E + 21$ n/cm² for all three irradiations, resulting in about 3 dpa in 316 stainless steel.

9.3 HYDROGEN GENERATION ARISING FROM THE ⁵⁹Ni(N,P) REACTION AND ITS IMPACT ON FISSION-FUSION CORRELATIONS – L. R. Greenwood and F. A. Garner (Pacific Northwest National Laboratory). 291

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10.1 SUMMARY OF RECOMMENDED CORRELATIONS FOR ITER-GRADE TYPE 316L(N) FOR THE ITER MATERIALS PROPERTIES HANDBOOK – M. C. Billone (Argonne National Laboratory) and J. E. Pawel (Oak Ridge National Laboratory). 295

The focus of this effort is the effects of irradiation on the ultimate tensile strength (UTS), the yield strength (YS), the uniform elongation (UE), the total elongation (TE) and the reduction in area (RA) in the ITER-relevant temperature range of 100-400°C. For the purposes of this summary, data for European heats of 316 with $0.02 \leq C \leq 0.03$ wt. % and $0.06 \leq N \leq 0.08$ wt. % are referred to as E316L(N) data and grouped together. Other heats of 316 and Ti-modified 316 are also included in the data base. For irradiation and postirradiation-test temperatures in the range of 200-400°C, the common behavior of these heats of stainless steel is a yield strength approaching the ultimate tensile strength, an ultimate tensile strength approaching 800 MPa, a uniform elongation approaching 0.3%, a total elongation approaching 3-9%, and a high (about 60%) reduction in area as the neutron damage approaches 10 dpa.

11.0 IRRADIATION FACILITIES, TEST MATRICES, AND EXPERIMENTAL METHODS. 305

11.1 SCHEDULE AND STATUS OF IRRADIATION EXPERIMENTS – A. F. Rowcliffe and M. L. Grossbeck (Oak Ridge National Laboratory). 307

The current status of reactor irradiation experiments is presented in tables summarizing the experimental objectives, conditions, and schedule.

11.2 STATUS OF DOE/JAERI COLLABORATIVE PROGRAM PHASE II AND PHASE III CAPSULES – J. E. Pawel, K. E. Lenox (Oak Ridge National Laboratory), and I. Ioka (Japan Atomic Energy Research Institute). 312

During this reporting period, the HFIR-MFE-RB-200J-1 and HFIR-MFE-RB-400J-1 spectrally tailored capsules were disassembled and the individual specimens recovered, sorted, and identified. Tensile testing and irradiation creep measurements will be performed during the next reporting period.

- 11.3 ATR-A1 IRRADIATION EXPERIMENT ON VANADIUM ALLOYS AND LOW-ACTIVATION STEELS – H. Tsai, R. V. Strain, I. Gomes, A. G. Hins, and D. L. Smith (Argonne National Laboratory), and H. Matsui (Tohoku University). 314

To study the mechanical properties of vanadium alloys under neutron irradiation at low temperatures, an experiment was designed and constructed for irradiation in the Advanced Test Reactor (ATR). The experiment contained Charpy, tensile, compact tension, TEM, and creep specimens of vanadium alloys. It also contained limited low-activation ferritic steel specimens as part of the collaborative agreement with Monbusho of Japan. The design irradiation temperatures for the vanadium alloy specimens in the experiment are ≈ 200 and 300°C , achieved with passive gas-gap sizing and fill-gas blending. To mitigate vanadium-to-chromium transmutation from the thermal neutron flux, the test specimens are contained inside gadolinium flux filters. All specimens are lithium-bonded. The irradiation started in Cycle 108A (December 3, 1995) and is expected to have a duration of three ATR cycles and a peak fluence of 4.5 dpa.

- 11.4 DISASSEMBLY OF IRRADIATED LITHIUM-BONDED CAPSULES CONTAINING VANADIUM ALLOY SPECIMENS – H. Tsai and R. V. Strain (Argonne National Laboratory). 321

Capsules containing vanadium alloy specimens from irradiation experiments in FFTF and EBR-II are being processed to remove the lithium bond and retrieve the specimens for testing. The work has progressed smoothly.

- 11.5 MICROSCOPIST'S AIDE: A COMPUTER PROGRAM WRITTEN TO ANALYZE TEM MICROGRAPHS – D. E. Reinhart (Graduate Research Assistant, University of Missouri, Rolla), and D. S. Gelles (Pacific Northwest National Laboratory). 322

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- 11.6 PASSIVE SiC IRRADIATION TEMPERATURE MONITOR – G. E. Youngblood (Pacific Northwest Laboratory). 324

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