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To explore the upper operating limit of vanadium-base alloys, SS-3 tensile specimens were prepared from the 832665 and 832864 heats of V-4Cr-4Ti alloys and tested at temperatures between 600 and 800°C. The results showed the behavior of the two heats to be similar (with the 832864 heat being slightly weaker) and the reduction of strengths with temperature insignificant at least up to 750°C. Ductility for both materials is good in the test temperature range. These findings are largely consistent with previously reported results on these two heats.

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1.3 IMPACT PROPERTIES OF GA'S 832864 HEAT OF V-4Cr-4Ti ALLOY AFTER EXPOSURE IN THE JFT-2M TOKAMAK – H. Tsai (Argonne National Laboratory), W. R. Johnson, P. W. Trester (General Atomics), S. Sengoku (Japan Atomic Energy Research Institute) 17

Four Charpy impact specimens were received from General Atomics (GA) after they were exposed in the tokamak environment of the JFT-2M. These specimens were prepared from the GA's 832864 heat of V-4Cr-4Ti material. The exposed specimens were impact tested at ANL using established procedures. The results showed no appreciable degradation of impact properties of the material from the JFT-2M exposure.

1.4 TENSILE PROPERTIES OF V-Cr-Ti ALLOYS AFTER EXPOSURE IN HYDROGEN-CONTAINING ENVIRONMENTS – K. Natesan and W. K. Soppet (Argonne National Laboratory) 21

A systematic study is underway at Argonne National Laboratory to evaluate the performance of several V-Cr-Ti alloys after exposure to environments containing hydrogen at various partial pressures. The goal is to correlate the chemistry of the exposure environment with hydrogen uptake by the samples and the resulting influence on the microstructures and tensile properties of the alloys. At present, the principal effort has focused on the V-4Cr-4Ti alloy of a heat identified as BL-71;

however, other alloys (V-5Cr-5Ti alloy of Heats BL-63 and T87, plus a V-4Cr-4Ti alloy from General Atomics [GA]) are also being evaluated. Other variables of interest are the effect of initial grain size on hydrogen uptake and tensile properties, and the synergistic effects of oxygen and hydrogen on the tensile behavior of the alloys. During this period, specimens of Heat BL-71 were exposed for 100 h to pH_2 in the range of 3×10^{-6} and 5×10^{-2} torr (2.7×10^{-5} to 0.45 Pa) at 200 and 350°C. Tensile test results obtained on specimens exposed for 100 h at 200, 350, and 500°C to pH_2 in the range of 1×10^{-6} to 0.01 torr, showed negligible effect of H_2 on either maximum engineering stress or uniform and total elongation. However, uniform and total elongation decreased substantially when the alloy was exposed at 500°C to 1.0 torr pH_2.

1.5 TENSILE AND IMPACT PROPERTIES OF A V-3.1Ti-0.5Si ALLOY AFTER IRRADIATION IN THE FAST FLUX TEST FACILITY – H. Tsai, L. J. Nowicki, T. S. Bray, M. C. Billone, and D. L. Smith (Argonne National Laboratory) 25

Tensile and Charpy specimens made of alloy BL-42 (with a nominal composition of V-3.1Ti-0.5Si) were irradiated in the Fast Flux Test Facility (FFTF) in Cycle 11 at 520 and 600°C to up to 51 dpa. Some of these specimens were retrieved from storage and tested in this reporting period. The interest of this material stems from recent findings that show binary alloys of this class appear to exhibit lower susceptibility to radiation-induced embrittlement than other V-base alloys in low-temperature irradiation (<≈400°C). The results of our tests show the BL-42 material to have good tensile properties and fair impact properties after the 520-600°C irradiation and that alloys of this class may warrant further development for improved low-temperature performance.

1.6 EFFECT OF IRRADIATION ON TOUGHNESS-TEMPERATURE CURVES IN V-4Cr-4Ti – E. Donahue, G. R. Odette, G. E. Lucas, And J. W. Sheckherd (University Of California, Santa Barbara); A. F. Rowcliffe (Oak Ridge National Laboratory) 32

Small, pre-cracked, 1/3-sized Charpy specimens of V-4Cr-4Ti were irradiated to a dose of 0.5 dpa at a temperature of 200 ± 8°C. The specimens were tested in three point bending under static conditions to measure effective toughness-temperature curves. Fracture processes were examined by using confocal microscopy and fracture reconstruction methods. It was found that this alloy undergoes normal stress-controlled cleavage transition below a characteristic temperature, and that irradiation hardening increased the transition temperature significantly. Shifts in the transition temperature imposed by irradiation hardening can be reasonably predicted by a simple equivalent yield stress model.

1.7 MICROSTRUCTURAL INHOMOGENEITIES IN V-4Cr-4Ti MECHANICAL PROPERTY SPECIMENS PREPARED FOR IRRADIATION EXPERIMENTS IN HFBR, BOR-60, AND HFIR – A. F. Rowcliffe and D. T. Hoelzer (Oak Ridge National Laboratory) 42

Significant variations exist between the microstructures of the various batches of mechanical property specimens prepared for the 500 kg heat of V-4Cr-4Ti. The development of banded grain structures is shown to be related to non-homogeneous distribution of Ti (OCN) particles (0.1-0.3 μ in diameter) that occur in some, but not all, plate products. Precipitation of Ti (OCN) will occur during the hot

extrusion of the ingot in regions where the dislocation density is high and where the temperature falls below the Ti (OCN) solvus at 1125°C. Investigation of the potential benefits of a homogenizing treatment at 1150°C is recommended before proceeding with additional fabrication of plate products from the extruded bar.

1.8 STRUCTURAL ANALYSIS OF Ti-OXYCARBONITRIDES IN V-Cr-Ti BASED ALLOYS – D. T. Hoelzer (Oak Ridge National Laboratory) 59

A study was conducted to determine the crystal structure of Ti-OCN particles which are commonly observed in V-Cr-Ti based alloys. The information obtained from this study will be used to gain a better understanding of the physical metallurgy of the V-Cr-Ti based alloys. The precipitates examined in this study were those that formed in a V-4Cr-4Ti alloy (S-40) plate) that was annealed at 1000°C for 2 hours. The precipitates are characterized as having a globular shaped morphology with some degree of interfacial faceting. The structural analysis indicated that these precipitates are consistent with the m3m crystal point group.

1.9 LASER-WELDED V-Cr-Ti ALLOYS: MICROSTRUCTURAL AND MECHANICAL PROPERTIES – K. Natesan, C. B. Reed, Z. Xu, and D. L. Smith (Argonne National Laboratory) 64

A systematic study has been in progress at Argonne National Laboratory to examine the use of YAG or CO_2 lasers to weld sheet materials of V-Cr-Ti alloys and to characterize the microstructural and mechanical properties of the laser-welded materials. During this report period, thirty-eight weldments were made with a YAG laser; the emphasis was on determining the optimal weld parameters to achieve deep penetration in the welds while eliminating porosities found previously. A preliminary assessment was then made of the weldments on the basis of visual appearance and sectioning to determine depth of penetration and extent of porosity.

1.10 DEVELOPMENT OF ELECTRICALLY INSULATING CaO COATINGS – K. Natesan, M. Uz, and S. Wieder (Argonne National Laboratory) 69

A systematic vapor transport study has been initiated to develop electrically insulating CaO coatings that are compatible with use in a liquid Li environment. Several experiments were conducted to study how the deposition of Ca on V-4Cr-4Ti substrate alloys is affected by variations in process temperature and time, and specimen location, surface preparation, and pretreatment. During this reporting period, a setup has been completed to measure the electrical resistivity of the coatings in air or in an inert gas environment as a function of temperature up to 750°C. Some preliminary data are presented.

2.0 SILICON CARBIDE COMPOSITE MATERIALS 73

2.1 SiC FIBER EVALUATIONS – G. E. Youngblood and R. H. Jones (Pacific Northwest National Laboratory) 75

The SiC fiber types selected for the JUPITER 14J irradiation experiment are presented together with the rationale for their selection. The fiber diameter variation

from fiber-to-fiber within a tow and along single filament lengths was assessed by image analysis and SEM for four SiC-based fibers: Hi-Nicalon, Hi-Nicalon Type S, Tyron SA, and Dow Sylramic. Because of the apparently characteristic fiber diameter variations observed across tows (6-22 μm) as well as along single filaments (up to ±1.0 μm/cm), it is not appropriate to use average fiber diameters to determine individual fiber strengths or fiber strength distributions. Generally, SiC fibers with larger diameters have lower tensile strengths. Therefore, a 3-parameter modified Wiebull analysis which includes a diameter dependence parameter is preferred over the simple 2-paramater Weibull analysis to properly describe SiC fiber strength distributions. The newly introduced diameter dependence parameter may be a sensitive measure of fiber quality or performance.

2.2 FABRICATION OF SiC/SiC COMPOSITES USING ADVANCED FIBERS – N. L. Vaughn, L. L. Snead, R. A. Lowden (Oak Ridge National Laboratory); A. Kohyama and Y. Katoh (Kyoto University); J. L. Bailey, J. J. Henry, and A. M. Williams (Oak Ridge National Laboratory) 81

FCVI SiC/SiC composites are being fabricated and will be incorporated in the upcoming 14J experiment in conjunction with the Jupiter and Crest Projects. Three different fiber compositions, three different fiber architectures, and two different interface thicknesses are being evaluated simultaneously. All of the fibrous preforms received a pyrolytic carbon surface coating between the fiber and matrix interface. Many of the non-irradiated samples have been characterized and trends examined.

2.3 METHODS FOR JOINING SILICON CARBIDE COMPOSITES FOR HIGH TEMPERATURE STRUCTURAL APPLICATIONS – C. A. Lewinsohn and R. H. Jones (Pacific Northwest National Laboratory), M. Singh (NASA Lewis Research Center), T. Shibayama (Hokkaido University), T. Hinoki, M. Ando, Y. Katoh, and A. Kohyama (Kyoto University) 87

Joining methods are required to allow affordable fabrication of large or complex SiC/SiC components for fusion energy systems. Previous analysis of the criteria for successful and functional joints indicate that reaction-formed and polymer-derived silicon carbide should be considered as candidate joint materials. Efforts have been initiated to investigate the issues involved with fabrication and durability of these joints. This report summarizes initial investigations of the long-term thermal stability of silicon carbide joints formed by a reaction-based approach. Results indicate that the joint may contain unreacted phases that react further during high-temperature exposure. These results, and their implications, must be confirmed by additional investigations.

2.4 DOSE RATE DEPENDENCE OF THE AMORPHIZATION OF SILICON CARBIDE – L. L. Snead, S. J.Zinkle, W. S.Eatherly, D. K. Hensley, N. L.Vaughn, J. W. Jones (Oak Ridge National Laboratory) 93

Single crystal silicon carbide (SiC) has been 2 MeV silicon ion irradiated in various irradiation temperature and ion flux ranges to measure the effect of these parameters on the critical dose for amorphization. The temperature and flux range for which amorphization was observed ranged from 80 to 400 K and 0.066 to 3×10^{-4}

dpa/s, respectively. The critical dose, D_{crit}, was found by locating the depth of the boundary between partially crystalline and fully amorphous material using dark field TEM from samples prepared in cross section. This depth was compared to the damage profile as calculated using the TRIM-96 code. The temperature dependence of D_{crit} is found to agree well with previously reported values, though new evidence suggests a defect species becoming mobile in the 250-300 K range. Also of significance is that D_{crit} was dependent on flux at 340 K, ranging from 0.79 displacements per atom at the lowest ion flux to ~0.6 dpa at the highest flux level. The dose rate dependence of D_{crit} is compared with a chemical rate theory model previously described by the authors. It is seen that the dose rate dependence is substantially weaker than theorized. An extrapolation of the measured dose rate dependence is also compared with recent data on fast neutron amorphized SiC.

2.5 INTERPHASE INTEGRITY OF NEUTRON IRRADIATED SILICON CARBIDE COMPOSITES – L. L. Snead and E. Lara-Curzio (Oak Ridge National Laboratory) 100

SiC/SiC composites were fabricated from Hi-Nicalon™ fibers with carbon, pseudo-porous SiC and multilayer SiC interphases. These materials were then irradiated in the High Flux Beam Reactor with fast neutrons to a dose level equivalent to 1.1 dpa. Results are presented for bend strength of both non-irradiated and irradiated materials. Degradation in ultimate bend stress was seen for all materials studied, while the matrix micro-cracking stress was unchanged. Within the interphases studied the multilayer SiC interphase material showed the least degradation (8-20%) in ultimate bend stress, while porous SiC underwent the greatest degradation (~35%). The fiber matrix interphases are studied with TEM for both non-irradiated and irradiated materials. While no irradiation induced microstructural evolution of the interphase was observed, debonding of the interphase from the fiber was observed for all cases. This debonding is attributed tensile stresses developed at the interface due to densification of the Hi-Nicalon™ fiber. Residual stress analysis of the fiber matrix interface clearly indicates that for densification of Hi-Nicalon™ and volumetric expansion of the CVD SiC matrix corresponding to these irradiation conditions tensile stresses occur well in excess of those which can be withstood by these, or any other viable SiC composite interphase.

3.0 FERRITIC/MARTENSITIC STEELS 109

3.1 SUMMARY OF IEA WORKSHOP/WORKING GROUP MEETING ON FERRITIC/MARTENSITIC STEELS FOR FUSION – R. L. Klueh (Oak Ridge National Laboratory) 111

The International Energy Agency (IEA) Working Group on Ferritic/Martensitic Steels for Fusion held a workshop at ECN Nuclear Research, Petten, The Netherlands, 1-2 October 1998. The Working Group, consisting of researchers from Japan, the European Union, the United States, and Switzerland, met to review research that has been completed since the previous meeting and to continue planning and coordinating an international collaborative test program on reduced-activation ferritic/martensitic steels for fusion applications. At the workshop, data were presented from the continuing research on the IEA heats of steel that are being studied in the collaboration. Data on these and other reduced-activation steels in the irradiated and unirradiated condition were presented. Other subjects that were

discussed included effects of a ferromagnetic steel in a fusion machine, the effect of helium on properties, and the development and application of oxide dispersion-strengthened steels for fusion. A Working Group status-review meeting is planned in conjunction with the International Conference on Fusion Reactor Materials (ICFRM-9) in Colorado Springs, 10-15 October 1999, at which time plans for a workshop to be held in 2000 will be finalized.

3.2 MECHANICAL PROPERTIES OF TWO 7-9Cr FERRITIC/MARTENSITIC STEELS – G .R. Odette, G. E. Lucas, and P. Spätig (University of California, Santa Barbara) 119

Tensile and fracture tests were performed on two ferritic/martensitic stainless steels. The temperature dependence of the yield stress and the thermal stress dependence of the activation volume were found to be in good agreement with a model based on the propagation of double kinks on screw dislocation segments. Effective fracture toughness-temperature curves were developed for two specimen sizes, and a constraint correction based on a critical stress (σ)-critical area (A) model was found to rationalize the differences. The status of the Master Curve Experiment is discussed.

3.3 EFFECT OF SMALL AMOUNTS OF RHENIUM AND OSMIUM ON MECHANICAL PROPERTIES OF A 9Cr-2W-0.25V-0.07Ta-0.1C Steel – R. L. Klueh, D. J. Alexander, and M. A. Sokolov (Oak Ridge National Laboratory) 125

The nuclear transmutation of tungsten to rhenium and osmium in a tungsten-containing steel irradiated in a fission or fusion reactor could change substantially the chemical composition of the steel. To determine the possible consequences of such changes on mechanical properties, tensile and Charpy impact properties were determined on five 9Cr-2W-0.25V-0.07Ta-0.1C steels that contained different amounts of rhenium, osmium, and tungsten. The mechanical properties changes due to these changes in composition were relatively minor. Observations were also made on the effect of carbon concentration. The effect of carbon on tensile behavior was relatively minor, but there was a large effect on Charpy properties. The steels showed relatively little effect of tempering temperature on the Charpy transition temperature, which was tentatively attributed to the silicon and/or manganese concentration.

3.4 SHEAR PUNCH TESTING OF IRRADIATED ISOTOPICALLY TAILORED FERRITIC/MARTENSITIC STEELS – M. L. Hamilton, D. S. Gelles (Pacific Northwest National Laboratory, S. Ohnuki (Hokkaido University), K. Shiba (JAERI), Y. Kohno (University of Tokyo), and A. Kohyama (Kyoto University) 136

Single variable experiments are being conducted to study effects of H/He/dpa on properties based on isotopically tailored alloys. ^{54}Fe has been used to prepare an isotopically tailored duplicate of the commercial steel F82H, and a small number of TEM disks have been irradiated in order to study radiation embrittlement. From single disk specimens, mechanical properties were obtained using a shear punch technique that produces a 1 mm blank from the 3 mm disk. Results indicate that shear punch testing can be used successfully to provide mechanical property data from single TEM disks. Little effect of helium on properties was found, either in ^{54}Fe isotopically tailored specimens or in a boron-doped specimen.

3.5 MICROSTRUCTURAL EXAMINATION OF IRRADIATED ISOTOPICALLY TAILORED FERRITIC/MARTENSITIC STEELS FOLLOWING SHEAR PUNCH TESTING – D. S. Gelles (Pacific Northwest National Laboratory), S. Ohnuki (Hokkaido University), K. Shiba (JAERI), Y. Kohno (University of Tokyo), A. Kohyama (Kyoto University), J. P. Robertson (Oak Ridge National Laboratory), and M. L. Hamilton (Pacific Northwest National Laboratory) 143

Single variable experiments are being conducted to study effects of H/He/dpa on properties based on isotopically tailored alloys. ^{54}Fe has been used to prepare an isotopically tailored duplicate of the commercial steel F82H, and a small number of TEM disks have been irradiated in order to study radiation embrittlement. From single disk specimens, mechanical properties were obtained using a shear punch technique that produces a 1 mm blank from the 3 mm disk. The 1 mm blanks have been thinned and examined by TEM. The novel thinning procedures are described and microstructural observation presented in detail. Little effect of helium on microstructure was found, either in ^{54}Fe isotopically tailored specimens or in a boron doped specimen.

3.6 HYDROGEN PRODUCTION EFFECT ON MICROSTRUCTURE OF HFIR-IRRADIATED LOW-ACTIVATION F82H STEEL USING ^{54}Fe ISOTOPE – E. Wakai (Japan Atomic Energy Research Institute), N. Hashimoto (Oak Ridge National Laboratory), K. Shiba and Y. Miwa (JAERI), and J. P. Robertson and R. L. Klueh (Oak Ridge National Laboratory) 151

Reduced-activation F82H-std and a similar F82H doped with ^{54}Fe (F82[(^{54}Fe]) steels were irradiated at 250°C to 2.8 dpa in HFIR and examined by TEM. The produced hydrogen concentration in the F82H(^{54}Fe) steel is estimated to be 68 appm. Pre-irradiation microstructures of F82H(^{54}Fe) steel were very similar to those of F82H-std steel. A few small cavities were formed in the irradiated F82H(^{54}Fe) steel, but the swelling is insignificant, while in the F82H-std steel no cavities were observed. Precipitates with contrast similar to α' phase were observed on many dislocation loops in these steels. The number density and mean size for dislocation loops in the F82H-std and Fu2H(^{54}Fe) steels are 1.4×10^{22} m^{-3} and 7.9 nm, and 2.1×10^{22} m^{-3} and 6.6 nm, respectively. These types of loops are **b**=(**a**/2)<111> for the F82H-std and **b**=(**a**/2)<111> and **a**<100> for the F82H(^{54}Fe) steel. The concentration of **a**<111> type to all loops in the F82H(^{54}Fe) steel is about 73%.

3.7 SWELLING OF THE HFIR-IRRADIATED F82H, F82H+ ^{10}B AND F82H+ ^{58}Ni STEELS – E. Wakai (Japan Atomic Energy Research Institute), N. Hashimoto (Oak Ridge National Laboratory), K. Shiba, and Y. Miwa (JAERI), and J. P. Robertson and R. L. Klueh (Oak Ridge National Laboratory) 161

Swelling of reduced-activation F82H-std and F82H steels doped with natural boron (311 appm), isotope ^{10}B (325 appm), 1.35 at% ^{58}Ni, and 1.31 at% ^{60}Ni irradiated at 300 and 400°C to 52 dpa in the HFIR have been examined by TEM. The swelling of F82H-std irradiated at 400°C to 52 dpa is about 0.6% and the natural B and ^{10}B doped F82H steels is about 0.9 and 1.1%, respectively. In the ^{58}Ni and ^{60}Ni doped F82H steels, swelling is 0.02 and 0%, respectively, even though the ^{58}Ni-doped specimen has the highest helium production. Large cavities in the F82H-std are observed in the matrix but not observed near many lath boundaries, while in the ^{10}B

doped specimens, cavities are formed even near lath boundaries. While the cavities formed at 300°C to 52 dpa are observed in only F82H+ ^{10}B and F82H+ ^{58}Ni steels, the swelling value is insignificant. The number densities of dislocation loops formed in these steels at 300°C to 52 dpa are very high (i.e., on the order of 10^{22} m^{-3}), and at 400°C to 52 dpa the number densities are very low (i.e., on the order of 10^{20} m^{-3}). A high density of precipitates is formed in the matrix of the Ni-doped F82H steels. The low swelling of Ni-doped specimens at 400°C of 52 dpa may be caused by the formation of a high density of precipitates.

3.8 MICROSTRUCTURE OF ISOTOPICALLY-TAILORED F82H AND HT9 IRRADIATED AT 400°C IN HFIR – N. Hashimoto, J. P. Robertson, A. F. Rowcliffe (Oak Ridge National Laboratory, Y. Miwa and K. Shiba (Japan Atomic Energy Research Institute) 170

Microstructures of reduced-activation ferritic/martensitic steels, F82H (8Cr-2W-0.2V-0.04Ta), 8Cr-2WVTa doped with 58Ni, HT9 (12Cr-1MoVW) and HT9 doped with ^{58}Ni, irradiated at 400°C to 7 dpa in the High Flux Isotope Reactor (HFIR), were investigated by transmission electron microscopy. Cavities were observed in the standard F82H (f82H-std.) and 8 Cr-2WVTa-^{58}Ni, but not in the standard HT9 (HT9-std.) and HT9-^{58}Ni alloys. The irradiation induced $\mathbf{a}_0$<100> and ($\mathbf{a}_0$/2)<111> type dislocation loops in all alloys; the number density and the mean diameter of ($\mathbf{a}_0$2)<111> type loops were lower and smaller than that of $\mathbf{a}_0$<100> type loops. Also, there is a tendency that the number density of loops in the F82H and 8Cr-2WVTa-^{58}Ni alloys were lower than those in the HT9 alloys.

3.9 SWELLING OF F82H IRRADIATED AT 673 K TO 7 dpa in HFIR – Y. Miwa, E. Wakai, K. Shiba (Japan Atomic Energy Research Institute), N. Hashimoto, J. P. Robertson, and A. F. Rowcliffe (Oak Ridge National Laboratory) 175

A reduced activation ferritic/martensitic steel F82H (F82H-std) and a heat with the addition of isotope ^{10}B (F82H+^{10}B) were irradiated at 673 K to 7 and 51 dpa in HFIR. The swelling behavior of these alloys was examined by transmission electron microscopy. In the F82H-std irradiated to 7 dpa, small cavities (~8 nm in dia) were observed in lath cells, but not on the interfaces of lath boundaries or precipitates. The cavity number density (N_c) was about 4×10^{21} m^{-3}. The swelling was about 0.2%. In the F82H+^{10}B irradiated to 7 dpa, smaller cavities (~5 nm in dia) were observed in lath cells, and some cavities occurred on the interfaces of lath boundaries or small precipitates. The N_c was about 1.5×10^{22} m^{-3}, and the swelling was about 0.2%. In the F82H std irradiated to 51 dpa, large and small cavities were observed in lath cells, but not on the interfaces of lath boundaries or precipitates. The N_c was about 1×10^{21} m^{-3}, and the swelling was about 0.6%.

4.0 COPPER ALLOYS AND HIGH HEAT FLUX MATERIALS 183

4.1 TEMPERATURE AND STRAIN RATE EFFECTS IN SOLUTION ANNEALED CuCrZr – D. J. Edwards (Pacific Northwest National Laboratory) 185

Previous reports presented the effects of strain rate and test temperature on the tensile properties of GlidCop™ Al25 and Hycon 3HP™ tested in air. The present report summarizes the results of a similar set of tests on solution annealed (SA)

CuCrZr. The SA CuCrZr proved to have only a slight dependence on test temperature up to 350°C in comparison to the other alloys tested in this series. Tests conducted at 450°C, irrespective of the strain rate used, produced a low level of precipitate aging in the solution annealed alloy that masked any potential effect of strain rate and test temperature. For a given strain rate the yield and ultimate strength decreased slightly up to 350°C, then increased significantly at 450°C as the aging began. The uniform and total elongation both decreased steadily as the test temperature was increased to 450°C. The reduction in area was less sensitive to temperature up to 250°C, then decreased steadily up to 450°C. In general the tensile results showed little dependence on strain rate with the exception that the reduction in area was somewhat higher at the two highest strain rates (1.5×10^{-1} and $1.5\ s^{-1}$) when tested at 350 and 450°C, which may be a consequence to the age hardening. Failure mode in the material appeared ductile throughout the range of strain rate and test temperatures.

5.0 AUSTENITIC STAINLESS STEELS 195

5.1 MICROSTRUCTURAL EVOLUTION OF COLD-WORKED AUSTENITIC STAINLESS STEELS IRRADIATED TO 17 dpa IN SPECTRALLY TAILORED EXPERIMENTS OF THE ORR AND HFIR AT 400°C – E. Wakai (Japan Atomic Energy Research Institute), N. Hashimoto and J. P. Robertson (Oak Ridge National Laboratory), and T. Sawai and A. Hishinuma (JAERI) 197

The microstructural evolution of cold-worked JPCA, 316R, C, K steels irradiated at 400°C in spectrally tailored experiments of the ORR and HFIR was investigated. The helium generation rates were about 12-16 appm He/dpa on the average up to 17.3 dpa. The number density and average diameter of dislocation loops in the steels had ranges of 3×10^{21} to $8 \times 10^{21}\ m^{-3}$ and 14.4 to 23.7 nm, respectively. Carbides were formed in all steels by the irradiation, and the number density and average diameter had ranges of 2×10^{21} to $1 \times 10^{22}\ m^{-3}$ and 3.4 to 17.7 nm, respectively. The number density and root mean cube of radius for cavities were 6×10^{21} to $2 \times 10^{22}\ m^{-3}$, and 1.2 to 2.4 nm, respectively, in these steels, and the swelling was 0.007 to 0.1% in these steels. JPCA-CW which has the lowest swelling shows the highest number density of carbides, while K-CW which has the highest swelling shows the lowest density of carbides.

6.0 INSULATING CERAMICS AND OPTICAL MATERIALS 203

6.1 EFFECT OF LOW TEMPERATURE ION IRRADIATION ON THE MICROSTRUCTURE OF NITRIDE CERAMICS – S. J. Zinkle, L. L. Snead, W. S. Eatherly, J. W. Jones, and D. K. Hensley (Oak Ridge National Laboratory) 205

Cross-section transmission electron microscopy was used to investigate the microstructure of polycrystalline silicon nitride (Si_3N_4) and aluminum nitride (AlN) following 2 MeV Si ion irradiation at 80 and 400 K up to a fluence of 4×10^{20} ions/m^2 (maximum damage of ~10 displacements per atom, dpa). A buried amorphous band was observed at both temperatures in Si_3N_4 in the region corresponding to the peaks in the implanted ion and displacement damage. From a comparison of Si_3N_4 specimens irradiated at different fluences, it is concluded that the amorphization is primarily controlled by the implanted Si concentration rather than the displacement

damage level. Si_3N_4 amorphization did not occur in regions well-separated from the implanted ions for doses up to at least 3 dpa at 80 K, whereas amorphization occurred in the ion implanted region (calculated Si concentration >0.01 at.%) for damage levels as low as ~0.6 dpa. The volumetric swelling associated with the amorphization of Si_3N_4 is <10%. Amorphization was not observed in any of the irradiated AlN specimens. A moderate density of small (~3 nm) defect clusters were observed in the crystalline damaged regions of both the Si3N4 and AlN specimens at both irradiation temperatures. Aligned network dislocations were also observed in the AlN specimen irradiated to high dose at 80 K.

6.2 PHONON SCATTERING RATE AND THERMAL CONDUCTIVITY REDUCTION DUE TO DISLOCATION LOOPS IN ALUMINA – D. P. White (Oak Ridge National Laboratory and Merrimack College) 212

The phonon scattering rate due to dislocation loops has been calculated. The dislocation loops are modeled as plate-like inclusions in the lattice. Using the calculated value of the phonon scattering relaxation time the reduction in the lattice thermal conductivity is estimated. It is found that for characteristic values of loop size and number density in irradiated alumina dislocation loops will produce a fractional reduction in the thermal conductivity of approximately 33 percent, assuming this is the only scattering mechanism other than intrinsic scattering.

7.0 SOLID BREEDING MATERIALS 217

No contributions.

8.0 RADIATION EFFECTS, MECHANISTIC STUDIES, AND EXPERIMENTAL METHODS 219

8.1 DEVELOPMENT OF SUPERPOSITION RULES FOR HARDENING IN ALLOYS CONTAINING MULTIPLE DEFECT POPULATIONS – G. R. Odette, G. E. Lucas, G. Tedeski, and B. D. Wirth (University of California, Santa Barbara) 221

In this study extensive computer simulations were carried out to calculate the yield strengths arising from the presence of two defect populations. The motion of an individual dislocation through a random obstacle field containing varying ratios of obstacles with two different barrier strengths (strong and weak) was computed based on equilibrium bowing of dislocation segments between adjacent obstacles. Yield stress was determined as the minimum stress necessary for the dislocation to successfully traverse the obstacle field. The results show that the superposition law is neither linear nor root sum square, but can be approximately represented by a superposition weighting parameter that is a simple analytical function of the individual strong and weak barrier strengths. Illustrations of the implications of this law are provided.

8.2 MECHANISMS OF PLASTIC AND FRACTURE INSTABILITIES OF FUSION STRUCTURAL MATERIALS – N. M. Ghoniem (University of California, Santa Barbara) 227

The main thrust of the current modeling effort is to identify the mechanisms which lead to localization of plastic deformation, the accelerated cavitation at grain

boundaries, the detrimental influence of helium transmutations, and the loss of ductile crack propagation in irradiated materials. We report here our initial research findings on the problem of accumulation of nano-scale prismatic defect clusters in the vicinity of dislocations. Under irradiation conditions, small defect clusters are produced in collision cascades. These mobile clusters interact with dislocations hindering its motion. Thus, radiation hardening takes place as a result of these interactions. However, it is shown here that once these nano-phase clusters approach dislocations, they may be trapped in their elastic strain fields. Trapping is shown to occur by elastic interactions within a zone of 10 nm in bcc Fe, and 20 nm in fcc Cu at RT. If the local stress (i.e. applied plus internal) is high, clusters are absorbed in the core of grown-in dislocations as a result of unbalanced moments, providing sufficient energy for rotation of their Burgers vectors in a zone of 2-3 nm in Fe. Near the dislocation core, sessile defect clusters in Cu are shown to convert to a glissile configuration.

8.3 EVALUATION OF FRACTURE TOUGHNESS, CONSTRAINT AND SIZE EFFECTS FROM CRITICAL CRACK TIP OPENING MEASUREMENTS USING CONFOCAL MICROSCOPY AND FRACTURE RECONSTRUCTION METHODS – G. R. Odette, G. E. Lucas, K. Edsinger, E. Donahue, and H. Rathbun (University of California, Santa Barbara) 243

This work demonstrates the use of confocal microscopy and fracture reconstruction methods to measure the critical crack tip opening displacement for several candidate fusion reactor structural materials – including both tempered martensitic steels and V-4Cr-4Ti – as well as other alloys for a variety of testing conditions and specimen sizes and configurations. Extensions of the method to regimes that deviate from small scale yielding, including stable crack growth and small specimens, based on local measures of toughness combined with simulations of crack tip fields are also described.

8.4 COMPARISON OF SWELLING AND IRRADIATION CREEP BEHAVIOR OF FCC-AUSTENITIC AND BCC-FERRITIC/MARTENSITIC ALLOYS AT HIGH NEUTRON EXPOSURE – F. A. Garner (Pacific Northwest National Laboratory), M. B. Toloczko (Washington State University), and B. H. Sencer (New Mexico Tech) 252

Extended Abstract

8.5 PROGRESS ON THE INTEGRATED FOCUS ON FUNDAMENTAL STUDIES – VANADIUM INITIATIVE – H. L. Heinisch (Pacific Northwest National Laboratory) 254

IFFS-VI was initiated at the JUPITER Workshop on Theory and Modeling for Fusion Materials held October 30, 1997, during ICFRM-8 in Sendai, Japan. The original initiative is described, participants and collaborations are identified, and recent progress and future plans are discussed.

8.6 SIMULATION OF DAMAGE EVOLUTION AND ACCUMULATION IN VANADIUM – E. Alonso, M.-J. Caturla, T. Díaz de la Rubia (Lawrence Livermore National Laboratory), and J.M. Perlado** (Instituto de Fusión Nuclear, José Gutiérrez Abascal, 2 E.T.S.I.I.) 259

Energetic atoms which have been knocked off their lattice sites by neutron or ion irradiation leave a trail of vacancies and interstitials in their wake. Most of these defects recombine or cluster within their own collision cascade. Some fraction, however, escape to become freely migrating defects (FMD) in the bulk of the material. The interaction of FMD with the microstructure has long been linked to changes in the macroscopic properties of materials under irradiation. We calculate the fraction of FMD in pure vanadium for a wide range of temperatures and primary knock-on atom (PKA) energies. The collision cascade database is obtained from molecular dynamics (MD) simulations with an embedded atom method (EAM) potential. The actual FMD calculation is carried out by a kinetic Monte Carlo (kMC) code with a set of parameters extracted either from the experimental literature or from MD simulations. We take two different approaches to the problem and compare them. The first consists of an idealized simulation for single cascades. Annealing each cascade at different temperatures allows the mobile species to escape and account for FMD. The second analyzes bulk diffusion and damage accumulation in a specimen irradiated at a low dose rate in the presence of impurities, in order to mimic experimental conditions. At the temperature studied, beginning of stage V, we observe that only vacancies are free to move whereas most interstitials are stopped by impurities. The fraction of FMD obtained is 11% for high purity vanadium, which is in good agreement with the figures reported in literature. We also analyze the role of impurities in damage accumulation.

8.7 KINETIC MONTE CARLO STUDIES OF THE EFFECTS OF ONE-DIMENSIONAL GLIDE ON THE REACTION KINETICS OF INTERSTITIAL CLUSTERS – H. L. Heinisch (Pacific Northwest National Laboratory), B. N. Singh (Risø National Laboratory), and S. I. Golubov (Institute of Physics and Power Engineering) 271

Extended abstract.

9.0 DOSIMETRY, DAMAGE PARAMETERS, AND ACTIVATION CALCULATIONS 275

No contributions.

10.0 MATERIALS ENGINEERING AND DESIGN REQUIREMENTS 277

10.1 MATERIALS INPUT FOR THE ARIES-ST DESIGN PROGRAM – M. C. Billone (Argonne National Laboratory) 279

The ARIES-ST design incorporates a number of materials for which data and design criteria are required. The materials input includes the present status of the materials database, as well as progress in fabrication techniques and materials development which can be reasonably achieved over the next 25-50 years. During the past year, input was provided for the copper alloys considered for the electrically-conducting center post, the aluminum alloys proposed for the toroidal field return outer shell, the ferritic steels (low-activation and ODS) to be used as first-wall and blanket

structural material, the compatibility between the slow-flowing Pb-Li breeder and the SiC thermal barrier and the ferritic steel structural material, and the tungsten selected for both the plasma-facing divertor material and the divertor coolant pipes. Some guidance was also provided with regard to fabrication and joining techniques. The level of detail provided with regard to the database and design guidelines for the various components depended on the functional importance of the component and the status of the database.

11.0 IRRADIATION FACILITIES, TEST MATRICES, AND EXPERIMENTAL METHODS 283

11.1 ACCELERATED HELIUM PRODUCTION IN Fe-54 DOPED ALLOYS – MEASUREMENTS AND CALCULATIONS FOR THE FIST (FERRITIC ISOTOPIC TAILORING) EXPERIMENT – L. R. Greenwood, B. M. Oliver, D. S. Gelles (Pacific Northwest National Laboratory), S. Ohnuki (Hokkaido University), K. Shiba (JAERI), Y. Kohno (University of Tokyo), A. Kohyama (Kyoto University), and J. P. Robertson (Oak Ridge National Laboratory) 285

Iron alloys enriched in Fe-54 were irradiated in the High Flux Isotopes Reactor (HFIR). Measurements and calculations have been performed to demonstrate and to predict the accelerated helium production due to higher fast and/or thermal neutron cross sections for helium-producing reactions on the iron isotopes of Fe-54 and Fe-55. A helium/dpa ratio of 2.3 was achieved for a 1.25-year irradiation.

11.2 MODULATED-BEAM MASS SPECTROMETRY SYSTEM FOR THE DETERMINATION OF LOW LEVELS OF HYDROGEN IN IRRADIATED MATERIALS – B. M. Oliver, J. A. Abrefah, F. A. Garner (Pacific Northwest National Laboratory), and A. Kumar (University of Missouri, Rolla) 291

A high-sensitivity analysis system has been developed for the measurement of hydrogen in solid materials. The system is based on a low-volume hot crucible hydrogen extraction furnace in combination with a modulated-beam quadrupole mass spectrometer detector system. Calibration of the system is conducted using a commercially available hydrogen leak source attached to the detector volume input line. Initial tests on the system using 5 to 10 mg sized specimens cut from a hydrogen-containing steel standard of 5.2 ± 0.3 wppm gave an average concentration of 5.8 ± 0.2 wppm. The ultimate system detection limit is currently ~5 $\times 10^{14}$ atoms, or ~5 appm for a 10 mg specimen. Future work will now involve analysis of a number of irradiated materials from various fission and fusion research programs.

11.3 CONCEPTUAL DEVELOPMENT OF THE FUSION-2 EXPERIMENT FOR IRRADIATION TESTING OF VANADIUM ALLOYS IN A LITHIUM ENVIRONMENT AT 500-700°C IN THE BOR-60 REACTOR – V. Kazakov, V. Chakin, V. Efimov, V. Petukhov, A. Tuktabiev, P. Gabiev (Research Institute of Atomic Reactors), H. Tsai, T. S. Bray, D. L. Smith (Argonne National Laboratory), and A. Rowcliffe (Oak Ridge Nationall Laboratory) 296

The specific requirements of this task were to complete the conceptual designs of irradiation capsules to a neutron dose of approximately 20 dpa in BOR-60. The

specimen matrix will include sheet tensile, compact tension, bend bars, irradiation creep tubes, and TEM disks. The irradiation temperatures will be approximately 500, 550, 600, and 650°C; the temperature changes related to changes in reactor operating conditions are to be minimized; and all specimens are to be encapsulated in lithium. In addition, the possibility of a multi-temperature assembly and reconstitution of assemblies was assessed.

11.4 PROGRESS REPORT ON THE OPERATION OF THE VARYING TEMPERATURE EXPERIMENT (HFIR-MFE-RB-13J) – A. L. Qualls (Oak Ridge National Laboratory) and T. Muroga (National Institute for Fusion Science) 302

Irradiation of the HFIR-MFE-RB-13J irradiation experiment began during this reporting period. Four cycles of operation were completed and the fifth cycle began during the reporting period. While the operating temperatures of the experiment are slightly higher than anticipated, control of the experiment has worked as expected including temperature control during reactor startup.

11.5 OPERATING HISTORY OF THE HFIR-MFE-RB-11J AND –12J (P3-2 AND P3-3) EXPERIMENTS – K. E. Lenox and M. L. Grossbeck (Oak Ridge National Laboratory) 307

The HFIR MFE-RB-11J and –12J experiments are being conducted under the auspices of the Japanese Monbusho, the Japan Atomic Energy Research Institute (JAER), and the U.S. Fusion Energy Sciences Program. The irradiation of the experiments was completed in July of 1998, and the disassembly of the experiments is scheduled to be complete in early 1999. The operating history of both experiments, including the operating temperatures of the specimens, is discussed.

11.6 AS-BUILT CONDITION AND OPERATING HISTORY OF THE U.S./JAERI HFIR MFE-RB-10J CAPSULE – K. E. Lenox and J. P. Robertson (Oak Ridge National Laboratory) 324

The HFIR-MFE-RB-10J experiment is being conducted under the auspices of the U.S. Department of Energy (DOE)/Japan Atomic Energy Research Institute (JAERI) Collaborative Testing Program. The final specimen loading lists, operating constraints, and operating history of the experiment are discussed. The experiment has undergone approximately ten (10) days of irradiation, but has been removed from HFIR due to the development of a leak in the primary containment vessel. The associated europium liner has also been removed. Efforts are underway to apply for approval to continue the irradiation of the RB-10J experiment as soon as possible.

11.7 STATUS OF DOE/JAERI COLLABORATIVE PROGRAM PHASE II AND PHASE III CAPSULES – J. P. Robertson, K.E. Lenox (Oak Ridge National Laboratory), and Y. Miwa (Japan Atomic Energy Research Institute) 334

Significant progress continues to be made in the post-irradiation examinations (PIE) of the specimens from the DOE ORNL/JAERI collaborative capsules and in the design and fabrication of additional capsules. This report serves as a summary of the irradiation parameters for the capsules and the Fusion Materials progress reports related to the design, loading, operation, and dosimetry.

11.8 SCHEDULE AND STATUS OF IRRADIATION EXPERIMENTS – A. F. Rowcliffe and J. P. Robertson (Oak Ridge National Laboratory) 337

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

11.9 PROGRESS REPORT ON THE TRIST-TC1 EXPERIMENT – A. L. Qualls, L. L. Snead, S. J. Zinkle, W. S. Eatherly, R.G. Sitterson, and D.W. Sparks (Oak Ridge National Laboratory), R. Yamada (Japan Atomic Energy Research Institute), and Y. Katoh (Kyoto University, Institute of Advanced Energy) 345

Fabrication of the experiment began during this reporting period. Forty specimens, in addition to eight heat generation measurement standards, were cut to size and brazed into twenty-four specimen pads. The specimen pads were assembled into three temperature zones comprising the experimental region of the capsule.