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Evaluation of trace element concentrations in vanadium alloys is important to characterize the low-activation characteristics and possible effects of trace elements on the properties. Detailed chemical analyses of several vanadium and vanadium alloy heats procured for the Argonne vanadium alloy development program were analyzed by Johnson-Matthey (UK) as part of a joint activity to evaluate trace element effects on the performance characteristics. These heats were produced by normal production practices for high grade vanadium. The analyses include approximately 60 elements analyzed in most cases by glow-discharge mass spectrometry. Values for molybdenum and niobium, which are critical for low-activation alloys, ranged from 0.4 to 60 wppm for the nine heats.

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V-4Cr-4Ti alloy has been selected for use in the manufacture of a portion of the DIII-D Radiative Divertor upgrade. The production of a 1200-kg ingot of V-4Cr-4Ti alloy, and processing into final sheet and rod product forms suitable for components of the DIII-D Radiative Divertor Program (RDP), has been completed by Wah Chang (formerly Teledyne Wah Chang) of Albany, Oregon (WCA). CVN impact tests on sheet material indicate that the material has properties comparable to other previously-processed V-4Cr-4Ti and V-5Cr-5Ti alloys. Joining of V-4Cr-4Ti alloy has been identified as the most critical fabrication issue for its use in the RDP, and research into several joining methods for fabrication of the RDP components, including resistance seam, friction, and electron beam welding, and explosive bonding is being pursued. Preliminary trials have been successful in the joining of V-alloy to itself by resistance, friction, and electron beam welding processes, and to Inconel 625 by friction welding. In addition, an effort to investigate the explosive bonding of V-4Cr-4Ti alloy to Inconel 625, in both tube-to-bar and sheet-to-sheet configurations, has been initiated, and results have been encouraging.

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The tensile data for all irradiated vanadium alloy samples and several unirradiated vanadium alloys tested at Argonne National Laboratory (ANL) have been critically reviewed and revised, as necessary. The review and revision are based on re-analyzing the original load-displacement strip-chart recordings using a methodology consistent with current ASTM standards. No significant difference has been found between the newly-revised and previously-reported values of yield strength (YS) and ultimate tensile strength (UTS). However, by correctly subtracting the non-gauge-length displacement and linear gauge-length displacement from the total cross-head displacement, the uniform elongation (UE) of the gauge length decreases by 4-9% strain and the total elongation (TE) of the gauge length decreases by 1-7% strain. These differences are more significant for lower-ductility irradiated alloys than for higher-ductility alloys.

- 1.4 TENSILE PROPERTIES OF VANADIUM ALLOYS IRRADIATED AT 390°C IN EBR-II — H. M. Chung, H.-C. Tsai, L. J. Nowicki, and D. L. Smith (Argonne National Laboratory) 18

Vanadium alloys were irradiated in Li-bonded stainless steel capsules to  $\approx 4$  dpa at  $\approx 390^\circ\text{C}$  in the EBR-II X-530 experiment. This report presents results of postirradiation tests of tensile properties of two large-scale (100 and 500 kg) heats of V-4Cr-Ti and laboratory (15-30 kg) heats of boron-doped V-4Cr-4Ti, V-8Cr-6Ti, V-5Ti, and V-3Ti-1Si alloys. Tensile specimens, divided into two groups, were irradiated in two different capsules under nominally similar conditions. The 500-kg heat (#832665) and the 100-kg heat (VX-8) of V-4Cr-4Ti irradiated in one of the subcapsules exhibited complete loss of work-hardening capability, which was manifested by very low uniform plastic strain. In contrast, the 100-kg heat of V-4Cr-4Ti irradiated in another subcapsule exhibited good tensile properties (uniform plastic strain 2.8-4.0%). A laboratory heat of V-3Ti-1Si irradiated in the latter subcapsule also exhibited good tensile properties. These results indicate that work-hardening capability at low irradiation temperatures varies significantly from heat to heat and is influenced by nominally small differences in irradiation conditions.

- 1.5 EFFECT OF HELIUM ON TENSILE PROPERTIES OF VANADIUM ALLOYS — H. M. Chung, M. C. Billone, and D. L. Smith (Argonne National Laboratory) 22

Tensile properties of V-4Cr-4Ti (Heat BL-47), 3Ti-1Si (BL-45), and V-5Ti (BL-46) alloys after irradiation in a conventional irradiation experiment and in the Dynamic Helium Charging Experiment (DHCE) were reported previously. This paper presents revised tensile properties of these alloys, with a focus on the effects of dynamically generated helium on ductility and work-hardening capability at  $< 500^\circ\text{C}$ . After conventional irradiation (negligible helium generation) at  $\approx 427^\circ\text{C}$ , a 30-kg heat of V-4Cr-4Ti (BL-47) exhibited very low uniform elongation, manifesting a strong susceptibility to loss of work-hardening capability. In contrast, a 15-kg heat of V-3Ti-1Si (BL-45) exhibited relatively high uniform elongation ( $\approx 4\%$ ) during conventional irradiation at  $\approx 427^\circ\text{C}$ , showing that the heat is resistant to loss of work-hardening capability.

Helium atoms produced at  $\approx 430^\circ\text{C}$  in dynamic helium charging irradiation seem to be conducive to higher ductility (compared to that under conventional irradiation) and relatively lower yield strength. This seemingly beneficial effect of helium is believed to be important in evaluating the performance of V-4Cr-4Ti and V-3Ti-1Si alloys, because susceptibility to loss of work-hardening capability at low temperatures under fusion-relevant helium-generating conditions is considered to be a major factor in governing the minimum operating temperature for fusion applications. In this respect, V-3Ti-1Si appears to be more advantageous than V-4Cr-4Ti, although other factors such as creep strength could be inferior. Tensile data from conventional irradiation experiments (i.e., negligible helium generation), especially the data for  $< 500^\circ\text{C}$ , appear to differ significantly from the results obtained with simultaneous helium generated by the DHCE. Therefore, a dynamic helium charging irradiation experiment is strongly recommended with a focus on determining tensile and fracture properties of V-4Cr-4Ti and V-3Ti-1Si alloys at 300-470°C at doses of 10-20 dpa with high helium/dpa ratios of  $\approx 4$ -5 appm He/dpa.

- 1.6 TENSILE PROPERTIES OF VANADIUM ALLOYS IRRADIATED AT 200°C IN THE HFIR — H. M. Chung, L. Nowicki, and D. L. Smith (Argonne National Laboratory) 29

Vanadium alloys were irradiated in a helium environment to  $\approx 10$  dpa at  $\approx 200^\circ\text{C}$  in the High Flux Isotope Reactor (HFIR). This report presents results of postirradiation

tests of tensile properties of laboratory heats of (V-(1-18)Ti, V-4Cr-4Ti, V-8Cr-6Ti, V-9Cr-5Ti, V-3Ti-1Si, and V-3Ti-0.1C alloys. Because of significant loss of work-hardening capability, all alloys except V-18Ti exhibited a very low uniform plastic strain of <1%. For V-Ti alloys, work-hardening capability increased with Ti content  $\geq 10\%$ , e.g., uniform strain of  $\approx 2.4\%$  for V-18Ti. The mechanism of the loss of work-hardening capability in the other alloys is not understood.

- 1.7 TENSILE PROPERTIES OF VANADIUM ALLOYS IRRADIATED AT  $<430^\circ\text{C}$  — H. M. Chung and D. L. Smith (Argonne National Laboratory) 33

Recent attention to vanadium alloys has focused on significant susceptibility to loss of work-hardening capability in irradiation experiments at  $<430^\circ\text{C}$ . An evaluation of this phenomenon was conducted on V-Ti, V-Cr-Ti, and V-Ti-Si alloys irradiated in several conventional and helium-charging irradiation experiments in the FFTF-MOTA, HFIR, and EBR-II. Work-hardening capability and uniform tensile elongation appear to vary strongly from alloy to alloy and heat to heat. A strong heat-to-heat variation has been observed in V-4Cr-4Ti alloys tested, i.e., a 500-kg heat (#832665), a 100-kg heat (VX-8), and a 30-kg heat (BL-47). The significant differences in susceptibility to loss of work-hardening capability from one heat to another are estimated to correspond to a difference of  $\approx 100^\circ\text{C}$  or more in minimum allowable operating temperature (e.g., 450 versus  $350^\circ\text{C}$ ).

- 1.8 MICROSTRUCTURAL EXAMINATION OF V-(4-5%)Cr-4-5%)Ti IRRADIATED IN X530 — D. S. Gelles (Pacific Northwest National Laboratory) and H. M. Chung (Argonne National Laboratory) 39

Microstructural examination results are reported for two heats of V-(4-5%)Cr-(4-5%)Ti irradiated in the X530 experiment to  $\sim 4$  dpa at  $\sim 400^\circ\text{C}$  to provide an understanding of the microstructural evolution that may be associated with degradation of mechanical properties. Fine precipitates were observed in high density intermixed with small defect clusters for all conditions examined following the irradiation. The irradiation-induced precipitation does not appear to be affected by preirradiation heat treatment at  $950$ - $1125^\circ\text{C}$ . There was no evidence for a significant density of large (diameter  $> 10$  nm) dislocation loops or network dislocations.

- 1.9 OXIDATION KINETICS AND MICROSTRUCTURE OF V-Cr-Ti ALLOYS EXPOSED TO OXYGEN-CONTAINING ENVIRONMENTS — K. Natesan (Argonne National Laboratory, M. Uz (Lafayette College, Easton, PA), and T. Ulie (Purdue University) 47

A systematic study is being conducted to determine the effects of time, temperature, and exposure environment on the oxidation behavior and microstructure of V-Cr-Ti alloys. All samples were from 1-mm-thick cold-rolled sheets, and each was annealed in vacuum at  $1050^\circ\text{C}$  for 1 h prior to high-temperature exposure. Different samples from each alloy were heated in air and low-oxygen environments at temperatures between 400 and  $650^\circ\text{C}$  for times up to a few hundred hours. Some exposures were conducted in a thermogravimetric analysis (TGA) apparatus, in which continuous measurements of weight change were recorded.

- 1.10 TENSILE PROPERTIES OF ALUMINIZED V-5Cr-5Ti ALLOY AFTER EXPOSURE IN AIR ENVIRONMENT — K. Natesan and W. K. Soppet (Argonne National Laboratory) 50

A pack diffusion process was used to enrich surface regions of V-5 wt.% Cr-5 wt.% Ti alloy specimens with aluminum, which is a more stable oxide former than vanadium; also, the oxide has a much slower growth rate than that of vanadium

oxide. Oxidation studies were conducted on surface-modified V-5Cr-5Ti alloy specimens in an air environment to evaluate the oxygen uptake behavior of the alloy as a function of temperature and exposure time. Uniaxial tensile tests were conducted at 500°C on several preoxidized specimens of the surface-modified alloy to examine the effects of oxidation and oxygen migration on tensile strength and ductility.

- 1.11 HEAT TREATMENT EFFECTS ON TENSILE PROPERTIES OF V-(4-5) wt.% Cr-(4-5) wt.% Ti ALLOYS — K. Natesan and W. K. Sopper (Argonne National Laboratory) 54

Effects of thermomechanical treatments on microstructures and mechanical properties are of interest for long term application of V-Cr-Ti alloys in fusion reactor systems. Influence of thermal annealing at 1050°C on stress/strain behavior, maximum engineering strength, and uniform and total elongation were evaluated. The results show that multiple annealing has minimal effect on the tensile properties of V-(4-5)Cr-(4-5)Ti alloys tested at room temperature at 500°C.

- 1.12 STUDY OF IRRADIATION CREEP OF VANADIUM ALLOYS — H. Tsai, R. V. Strain, and D. L. Smith (Argonne National Laboratory), and M. L. Grossbeck (Oak Ridge National Laboratory) 57

Thin-wall tubing was produced from the 832665 (500 kg) heat of V-4 wt.% Cr-4 wt.% Ti study its irradiation creep behavior. The specimens, in the form of pressurized capsules, were irradiated in Advanced Test Reactor and High Flux Isotope Reactor experiments (ATR-A1 and HFIR RB-12J, respectively). The ATR-A1 irradiation has been completed and specimens from it will soon be available for postirradiation examination. The RB-12J irradiation is not yet complete.

- 1.13 RECENT PROGRESS ON GAS TUNGSTEN ARC WELDING OF VANADIUM — M. L. Grossbeck, J. F. King, D. J. Alexander, and G. M. Goodwin (Oak Ridge National Laboratory) 61

Emphasis has been placed on welding 6.4 mm plate, primarily by gas tungsten arc (GTA) welding. The weld properties were tested using blunt notch Charpy testing to determine the ductile to brittle transition temperature (DBTT). Erratic results were attributed to hydrogen and oxygen contamination of the welds. An improved gas clean-up system was installed on the welding glove box and the resulting high purity welds had Charpy impact properties similar to those of electron beam welds with similar grain size. A post-weld heat treatment (PWHT) of 950°C for two hours did not improve the properties of the weld in cases where low concentrations of impurities were attained. Further improvements in the gas clean-up system are needed to control hydrogen contamination.

## 2.0 SILICON CARBIDE COMPOSITE MATERIALS 67

- 2.1 INVESTIGATION OF REACTIVITY BETWEEN SiC and Nb-1Zr IN PLANNED IRRADIATION CREEP EXPERIMENTS — C. A. Lewinsohn (Associated Western Universities), M. L. Hamilton and R. H. Jones (Pacific Northwest National Laboratory) 69

Thermodynamic calculations and diffusion couple experiments showed that SiC and Nb-1Zr were reactive at the upper range of temperatures anticipated in the planned irradiation creep experiment. Sputter-deposited aluminum oxide ( $Al_2O_3$ ) was selected as a diffusion barrier coating. Experiments showed that although the coating coarsened at high temperature it was an effective barrier for diffusion of

silicon from SiC into Nb-1Zr. Therefore, to avoid detrimental reactions between the SiC composite and the Nb-1Zr pressurized bladder during the planned irradiation creep experiment, a coating of Al<sub>2</sub>O<sub>3</sub> will be required on the Nb-1Zr bladder.

- 2.2 ANALYSIS OF NEUTRON IRRADIATION EFFECTS ON THERMAL CONDUCTIVITY OF SiC-BASED COMPOSITES AND MONOLITHIC CERAMICS — G. E. Youngblood and D. J. Senior (Pacific Northwest National Laboratory) 75

After irradiation of a variety of SiC-based materials to 33 or 43 dpa-SiC at 1000°C, their thermal conductivity values were degraded and became relatively temperature independent, which indicates that the thermal resistivity was dominated by point defect scattering. The magnitude of irradiation-induced conductivity degradation was greater at lower temperatures and typically was larger for materials with higher unirradiated conductivity. From these data, a  $K_{irr}/K_{unirr}$  ratio map which predicts the expected equilibrium thermal conductivity for most SiC-based materials as a function of irradiation temperature was derived. Due to a short-term EOC irradiation at 575° ± 60°C, a duplex irradiation defect structure was established. Based on an analysis of the conductivity and swelling recovery after post-irradiation anneals for these materials with the duplex defect structure, several consequences for irradiating SiC at temperatures of 1000°C or above are given. In particular, the thermal conductivity degradation in the fusion relevant 800-1000°C temperature range may be more severe than inferred from SiC swelling behavior.

- 2.3 CREEP BEHAVIOR FOR ADVANCED POLYCRYSTALLINE SiC FIBERS — G. E. Youngblood and R. H. Jones (Pacific Northwest National Laboratory), G. N. Morscher (Case Western Reserve University), and Akira Kohyama (Institute of Advance Energy, Kyoto University, Japan) 81

A bend stress relaxation (BSR) test is planned to examine irradiation enhanced creep in polycrystalline SiC fibers which are under development for use as fiber reinforcement in SiC/SiC composite. Baseline 1 hr and 100 hr BSR thermal creep "m" curves have been obtained for five selected advanced SiC fiber types and for standard Nicalon CG fiber. The transition temperature, that temperature where the S-shaped m-curve has a value 0.5, is a measure of fiber creep resistance. In order of decreasing thermal creep resistance, with the 100 hr BSR transition temperature given in parentheses, the fibers ranked: Sylramic (1261°C), Nicalon S (1256°C), annealed Hi Nicalon (1215°C), Hi Nicalon (1078°C), Nicalon CG (1003°C), and Tyranno E (932°C). The thermal creep for Sylramic, Nicalon S, Hi Nicalon, and Nicalon CG fibers in a 5000 hr irradiation creep BSR test is projected from the temperature dependence of the m-curves determined during 1 and 100 hr BSR control tests.

- 2.4 DESIGN OF A CREEP EXPERIMENT FOR SiC/SiC COMPOSITES IN HFIR — S. L. Hecht (Duke Engineering Hanford), M. L. Hamilton, R. H. Jones, G. E. Youngblood, and R. A. Schwartz (Pacific Northwest National Laboratory), and C. A. Lewinsohn (Associated Western Universities) 87

A new specimen was designed for performing in-reactor creep tests on composite materials, specifically on SiC/SiC composites. The design was tailored for irradiation at 800°C in a HFIR RB\* position. The specimen comprises a composite cylinder loaded by a pressurized internal bladder that is made of Nb1Zr. The experiment was designed for approximately a one year irradiation.

- 2.5 THERMOMECHANICAL INSTABILITY EFFECTS IN SiC-BASED FIBERS AND SiC<sub>f</sub>/SiC COMPOSITES — G. E. Youngblood, C. H. Henager, and R. H. Jones (Pacific Northwest National Laboratory) 111

Thermomechanical instability in irradiated SiC-based fibers with an amorphous silicon oxycarbide phase leads to shrinkage and mass loss. SiC<sub>f</sub>/SiC composites made with these fibers also exhibit mass loss as well as severe mechanical property degradation when irradiated at 800°C, a temperature much below the generally accepted 1100°C threshold for thermochemical degradation alone. The mass loss is due to an internal oxidation mechanism within these fibers which likely degrades the carbon interphase as well as the fibers in SiC<sub>f</sub>/SiC composites even in so-called "inert" gas environments. Furthermore, the mechanism must be accelerated by the irradiation environment.

- 3.0 FERRITIC/MARTENSITIC STEELS 119

- 3.1 HEAT TREATMENT EFFECTS ON IMPACT TOUGHNESS OF 9Cr-1MoVNb and 12Cr-1MoVW Steels Irradiated to 100 dpa — R. L. Klueh and D. J. Alexander (Oak Ridge National Laboratory) 121

Plates of 9Cr-1MoVNb and 12Cr-1MoVW steels were given four different heat treatments: two normalizing treatments were used and for each normalizing treatment two tempers were used. Miniature Charpy specimens from each heat treatment were irradiated to ≈19.5 dpa at 365°C and to ≈100 dpa at 420°C in the Fast Flux Test Facility (FFTF). In previous work, the same materials were irradiated to 4-5 dpa at 365°C and 35-36 dpa at 420°C in FFTF. The tests indicated that prior austenite grain size, which was varied by the different normalizing treatments, had a significant effect on impact behavior of the 9Cr-1MoVNb but not on the 12Cr-1MoVW. Tempering treatment had relatively little effect on the shift in DBTT for both steels. Conclusions are presented on how heat treatment can be used to optimize impact properties.

- 3.2 NEUTRON IRRADIATION EFFECTS ON THE DUCTILE-BRITTLE TRANSITION OF FERRITIC/ MARTENSITIC STEELS — R. L. Klueh and D. J. Alexander (Oak Ridge National Laboratory) 129

Extended abstract.

- 3.3 MICROSTRUCTURAL CHARACTERIZATION OF 5-9% CHROMIUM REDUCED-ACTIVATION STEELS — R. Jayaram (University of Pittsburgh) and R. L. Klueh (Oak Ridge National Laboratory) 130

The microstructures of a 9Cr-2W-0.25V-0.1C (9Cr-2WV), a 9Cr-2W-0.25V-0.07Ta-0.1C (9Cr-2WVTa), a 7Cr-2W-0.25V-0.07Ta-0.1C (7Cr-2WVTa), and a 5Cr-2W-0.25V-0.07Ta-0.1C (5Cr-2WVTa) steel (all compositions are in weight percent) have been characterized by Analytical Electron Microscopy (AEM) and Atom Probe Field Ion Microscopy (APFIM). The matrix in all four reduced-activation steels was 100% martensite. In the two 9Cr steels, the stable precipitates were blocky M<sub>23</sub>C<sub>6</sub> and small spherical MC. The two lower-chromium steels contained blocky M<sub>7</sub>C<sub>3</sub> and small needle-shaped carbonitrides in addition to M<sub>23</sub>C<sub>6</sub>. AEM and APFIM analysis revealed that in the steels containing tantalum, the majority of the tantalum was in solid solution. The experimental observations were in good agreement with phases and compositions predicted by phase equilibria calculations.

## 4.0 COPPER ALLOYS AND HIGH HEAT FLUX MATERIALS 141

### 4.1 EFFECT OF HEAT TREATMENTS ON THE TENSILE AND ELECTRICAL PROPERTIES OF HIGH-STRENGTH, HIGH-CONDUCTIVITY COPPER ALLOYS — S. J. Zinkle and W. S. Eatherly (Oak Ridge National Laboratory) 143

The unirradiated tensile properties of CuCrZr produced by two different vendors have been measured following different heat treatments. Room temperature electrical resistivity measurements were also performed in order to estimate the thermal conductivity of these specimens. The thermomechanical conditions studied included solution quenched, solution quenched and aged (ITER reference heat treatment), simulated slow HIP thermal cycle ( $\sim 1^\circ\text{C}/\text{min}$  cooling from solutionizing temperature) and simulated fast HIP thermal cycle ( $\sim 100^\circ\text{C}/\text{min}$  cooling from solutionizing temperature). Specimens from the last two heat treatments were tested in both the solution-cooled condition and after subsequent precipitate aging at  $475^\circ\text{C}$  for 2 h. Both of the simulated HIP thermal cycles caused a pronounced decrease in the strength and electrical conductivity of CuCrZr. The tensile and electrical properties were unchanged by subsequent aging in the slow HIP thermal cycle specimens, whereas the strength and conductivity following aging in the fast HIP thermal cycle improved to  $\sim 65\%$  of the solution quenched and aged CuCrZr values. Limited tensile and electrical resistivity measurements were also made on two new heats of Hycon 3HP CuNiBe. High strength but poor uniform and total elongations were observed at  $500^\circ\text{C}$  on one of these new heats of CuNiBe, similar to that observed in other heats.

### 4.2 INVESTIGATION OF THE INFLUENCE OF GRAIN BOUNDARY CHEMISTRY, TEST TEMPERATURE, AND STRAIN RATE ON THE FRACTURE BEHAVIOR OF ITER COPPER ALLOYS — K. Leedy and J. F. Stubbins (University of Illinois), D. J. Edwards (Pacific Northwest National Laboratory), R. R. Solomon (OMG Americas), and D. Kurs (Brush Wellman) 149

In an effort to understand the mechanical behavior at elevated temperatures ( $>200^\circ\text{C}$ ) of the various copper alloys being considered for use in the ITER first wall, divertor, and limiter, a collaborative study has been initiated by the University of Illinois and PNNL with two industrial producers of copper alloys, Brush Wellman and OMG Americas. Details of the experimental matrix and test plans have been finalized and the appropriate specimens have already been fabricated and delivered to the University of Illinois and PNNL for testing and analysis. The experimental matrix and testing details are described in this report.

## 5.0 AUSTENITIC STAINLESS STEELS 157

### 5.1 MICROSTRUCTURAL EVOLUTION OF AUSTENITIC STAINLESS STEELS IRRADIATED TO 17 dpa IN SPECTRALLY TAILORED EXPERIMENT OF THE ORR AND HFIR AT $400^\circ\text{C}$ — E. Wakai (Japan Atomic Energy Research Institute), N. Hashimoto (Oak Ridge National Laboratory), T. Sawai (JAERI), J. P. Robertson and L. T. Gibson (ORNL), I. Ioka and A. Hishinuma (JAERI) 159

The microstructural evolution of austenitic JPCA aged and solution annealed JPCA, 316R, C, K, and HP steels irradiated at  $400^\circ\text{C}$  in spectrally tailored experiments of the ORR and HFIR has been investigated. The helium generation rates were about 12-16 appm He/dpa on the average up to 17.3 dpa. The number densities and average diameters of dislocation loops in the steels have ranges of  $3.3 \times 10^{21}$ -  $9.5 \times 10^{21} \text{ m}^{-3}$  and 15.2- 26.3 nm, respectively, except for HP steel for which they are  $1.1 \times 10^{23} \text{ m}^{-3}$  and 8.0 nm. Precipitates are formed in all steels except for HP steel, and the values have ranges of  $5.2 \times 10^{20}$ -  $7.7 \times 10^{21} \text{ m}^{-3}$  and 3.4- 19.3 nm,

respectively. In the 316R, C, and K steels, the precipitates are also formed at grain boundaries, and the mean sizes are about 110, 50, and 50 nm, respectively. The number densities of cavities are about  $1 \times 10^{22} \text{ m}^{-3}$  in all the steels. The swelling is low in the steels which form the precipitates.

- 5.2 THE DEVELOPMENT OF A TENSILE-SHEAR PUNCH CORRELATION FOR YIELD PROPERTIES OF MODEL AUSTENITIC ALLOYS — G. L. Hankin (IPTME, Loughborough University), M. L. Hamilton and F. A. Garner (Pacific Northwest National Laboratory), and R. G. Faulkner (IPTME, Loughborough University) 169

The effective shear yield and maximum strengths of a set of neutron-irradiated, isotopically tailored austenitic alloys were evaluated using the shear punch test. The dependence on composition and neutron dose showed the same trends as were observed in the corresponding miniature tensile specimen study conducted earlier. A single tensile shear punch correlation was developed for the three alloys in which the maximum shear stress or Tresca criterion was successfully applied to predict the slope. The correlation will predict the tensile yield strength of the three different austenitic alloys tested to within  $\pm 53$  MPa. The accuracy of the correlation improves with increasing material strength, to within  $\pm 43$  MPa for predicting tensile yield strengths in the range of 400 to 800 MPa.

6.0 INSULATING CERAMICS AND OPTICAL MATERIALS 177

- 6.1 SUMMARY OF THE 9TH IEA WORKSHOP ON RADIATION EFFECTS IN CERAMIC INSULATORS — S. J. Zinkle (Oak Ridge National Laboratory), E. R. Hodgson (CIEMAT), and T. Shikama (Tohoku University) 179

Twenty one scientists attended an IEA workshop in Cincinnati, Ohio on May 8-9, 1997, which was mainly devoted to reviewing the current knowledge base on the phenomenon of radiation induced electrical degradation in ceramic insulators. Whereas convincing evidence for bulk RIED behavior has been observed by two research groups in sapphire after electron irradiation, definitive levels of bulk RIED have not been observed in high purity  $\text{Al}_2\text{O}_3$  by several research groups during energetic ion or fission neutron irradiation. Possible reasons for the conflicting RIED results obtained by different research groups were discussed. It was concluded that RIED does not appear to be of immediate concern for near-term fusion devices such as ITER. However, continued research on the RIED phenomenon with particular emphasis on electron irradiations of single crystal alumina was recommended in order to determine the underlying physical mechanisms. This will allow a better determination of whether RIED might occur under any of the widely varying experimental conditions in a fusion energy device. Several critical issues which are recommended for future study were outlined by the workshop attendees.

- 6.2 ANALYSIS OF IN-SITU ELECTRICAL CONDUCTIVITY DATA FROM THE HFIR TRIST-ER1 EXPERIMENT — S. J. Zinkle, L. L. Snead and W. S. Eatherly (Oak Ridge National Laboratory), E. H. Farnum (Los Alamos National Laboratory), T. Shikama (Tohoku University), and K. Shiiyama (Kyushu University) 188

The current vs. applied voltage data generated from the HFIR TRIST-ER1 experiment have been analyzed to determine the electrical conductivity of the 15 aluminum oxide specimens and the MgO-insulated electrical cables as a function of irradiation dose. With the exception of the 0.05%Cr-doped sapphire (ruby) specimen, the electrical conductivity of the alumina specimens remained at the expected radiation induced conductivity (RIC) level of  $<10^{-6} \text{ S/m}$  during full-power

reactor irradiation (10-16 kGy/s) at 450-500°C up to a maximum dose of ~3 dpa. The ruby specimen showed a rapid initial increase in conductivity to  $\sim 2 \times 10^{-4}$  S/m after ~0.1 dpa, followed by a gradual decrease to  $< 1 \times 10^{-6}$  S/m after 2 dpa. Nonohmic electrical behavior was observed in all of the specimens, and was attributed to preferential attraction of ionized electrons in the capsule gas to the unshielded low-side bare electrical leads emanating from the subcapsules. The electrical conductivity was determined from the slope of the specimen current vs. voltage curve at negative voltages, where the gas ionization effect was minimized. Dielectric breakdown tests performed on unirradiated mineral-insulated coaxial cables identical to those used in the HFIR TRIST-ER1 experiment indicate that the electrical shorting which occurred in many of the high voltage coaxial cables during the 3-month irradiation is attributable to thermal dielectric breakdown in the glass seals at the end of the cables, as opposed to a radiation-induced electrical degradation (RIED) effect.

- 6.3 IRRADIATION SPECTRUM AND IONIZATION-INDUCED DIFFUSION EFFECTS IN CERAMICS — S. J. Zinkle (Oak Ridge National Laboratory) 204
- There are two main components to the irradiation spectrum which need to be considered in radiation effects studies on nonmetals, namely the primary knock-on atom energy spectrum and ionizing radiation. The published low-temperature studies on  $\text{Al}_2\text{O}_3$  and  $\text{MgO}$  suggest that the defect production is nearly independent of the average primary knock-on atom energy, in sharp contrast to the situation for metals. On the other hand, ionizing radiation has been shown to exert a pronounced influence on the microstructural evolution of both semiconductors and insulators under certain conditions. Recent work on the microstructure of ion-irradiated ceramics is summarized, which provides evidence for significant ionization-induced diffusion. Polycrystalline samples of  $\text{MgO}$ ,  $\text{Al}_2\text{O}_3$ , and  $\text{MgAl}_2\text{O}_4$  were irradiated with various ions ranging from 1 MeV  $\text{H}^+$  to 4 MeV  $\text{Zr}^+$  ions at temperatures between 25 and 650°C. Cross-section transmission electron microscopy was used to investigate the depth-dependent microstructure of the irradiated specimens. Dislocation loop nucleation was effectively suppressed in specimens irradiated with light ions, whereas the growth rate of dislocation loops was enhanced. The sensitivity to irradiation spectrum is attributed to ionization-induced diffusion. The interstitial migration energies in  $\text{MgAl}_2\text{O}_4$  and  $\text{Al}_2\text{O}_3$  are estimated to be  $\leq 0.4$  eV and  $\leq 0.8$  eV, respectively for irradiation conditions where ionization-induced diffusion effects are expected to be negligible.
- 6.4 DEFECT PRODUCTION IN CERAMICS — S. J. Zinkle (Oak Ridge National Laboratory) and C. Kinoshita (Kyushu University) 211
- Extended abstract.
- 7.0 SOLID BREEDING MATERIALS 213
- No contributions.
- 8.0 RADIATION EFFECTS, MECHANISTIC STUDIES, AND EXPERIMENTAL METHODS 215
- 8.1 EFFECTS OF IN-CASCADE CLUSTERING ON NEAR-TERM DEFECT EVOLUTION — H. L. Heinisch (Pacific Northwest National Laboratory) 217

The effects of in-cascade defect clustering on the nature of the subsequent defect population are being studied using stochastic annealing simulations applied to

cascades generated in molecular dynamic (MD) simulations. The results of the simulations illustrate the strong influence of the defect configuration existing in the primary damage state on subsequent defect evolution. The large differences in mobility and stability of vacancy and interstitial defects and the rapid one-dimensional diffusion of small, glissile interstitial loops produced directly in cascades have been shown to be significant factors affecting the evolution of the defect distribution. In recent work, the effects of initial cluster sizes appear to be extremely important.

- 8.2 INFLUENCE OF SUBCASCADE FORMATION ON DISPLACEMENT DAMAGE AT HIGH PKA ENERGIES — R. E. Stoller (Oak Ridge National Laboratory) and L. R. Greenwood (Pacific Northwest National Laboratory) 221

Extended abstract.

- 9.0 DOSIMETRY, DAMAGE PARAMETERS, AND ACTIVATION CALCULATIONS 225

- 9.1 ANALYSIS OF THE DHCE EXPERIMENT IN THE POSITION A10 OF THE ATR REACTOR — I. C. Gomes, D. L. Smith, and H. Tsai (Argonne National Laboratory) 227

Calculations were performed to assess the possibility of performing DHCE experiments in mixed spectrum fission reactors. Calculated values of key parameters were compared with limit values for each quantity. The values calculated were: He-4 production from the  $6\text{Li}(n,t)4\text{He}$  reaction, tritium leakage, required tritium concentration in lithium, initial tritium charge per capsule, and helium to dpa ratio after 10 dpa of irradiation.

- 9.2 NEUTRONICS ANALYSIS OF THE DHCE EXPERIMENT IN ATR-ITV — I. C. Gomes, D. L. Smith, and H. Tsai (Argonne National Laboratory) 232

The preliminary analysis of the DHCE experiment in the ITV of ATR was performed and it was concluded that such a vehicle is suitable for this kind of experiment. It is recommended to place an extra filter material in the thermocouple sleeve (such as B-10), to improve the helium to dpa ratio profile during irradiation. Also, it was concluded that a preliminary estimation of period of time for replacement of the external filter would be around 5 dpa's.

- 10.0 MATERIALS ENGINEERING AND DESIGN REQUIREMENTS 239

No contributions.

- 11.0 IRRADIATION FACILITIES, TEST MATRICES, AND EXPERIMENTAL METHODS 241

- 11.1 PROGRESS REPORT ON THE VARYING TEMPERATURE EXPERIMENT — A. L. Qualls, M. T. Hurst, D. G. Raby, D. W. Sparks (Oak Ridge National Laboratory), and T. Muroga (National Institute for Fusion Science) 243

No summary provided.

- 11.2 THE MONBUSHO/U.S. SHIELDED HFIR IRRADIATION EXPERIMENT: HFIR-MFE-RB-11J AND 12J (P3-2 AND P3-3) — M. L. Grossbeck and K. E. Lenox (Oak Ridge National Laboratory, and T. Muroga (National Institute for Fusion Science) 254

This experiment is a joint project between Japanese Monbuscho and the U.S. Fusion Energy Sciences Program. It is the first of a series of experiments using europium oxide as a thermal neutron shield to minimize transmutations in vanadium alloys and ferritic-martensitic steels. The europium oxide shields were developed using ceramic processing techniques culminating in cold pressing and sintering. This experiment, which is a prototype for future fast neutron experiments in the HFIR, contains approximately 3200 specimens fabricated from 17 alloy types. The experiment began operating at 300 and 500°C in February 1997 and is projected to attain its goal fluence of ~5 dpa in February 1998.

- 11.3 NEUTRON IRRADIATION OF V-Cr-Ti ALLOYS IN THE BOR-60 FAST REACTOR: DESCRIPTION OF THE FUSION-1 EXPERIMENT — A. F. Rowcliffe (Oak Ridge National Laboratory), H. C. Tsai and D. L. Smith (Argonne National Laboratory), and V. Kazakov, and V. Chakin (RIAR, Dimitrovgrad) 284

The FUSION-1 irradiation capsule was inserted in Row 5 of the BOR-60 fast reactor in June 1995. The capsule contains a collaborative RF/U.S. experiment to investigate the irradiation performance of V-Cr-Ti alloys in the temperature range 310 to 350°C. This report describes the capsule layout, specimen fabrication history, and the detailed test matrix for the U.S. specimens. A description of the operating history and neutronics will be presented in the next semiannual report.

- 11.4 DESIGN CONSIDERATIONS OF THE IRRADIATION TEST VEHICLE FOR THE ADVANCED TEST REACTOR — H. Tsai, C. Gomes, and D. L. Smith (Argonne National Laboratory), A. J. Palmer, S. J. Hafer, and F. W. Ingram (Lockheed Martin Idaho Technologies Company), M. L. Hamilton (Pacific Northwest National Laboratory), K. R. Thoms (Oak Ridge National Laboratory), and F. W. Wiffen (U.S. Department of Energy) 300

An irradiation test vehicle (ITV) for the Advanced Test Reactor (ATR) is being jointly developed by the Lockheed Martin Idaho Technologies Company (LMIT) and the U.S. Fusion Program. The vehicle is intended for neutron irradiation testing of candidate structural materials, including vanadium-based alloys, silicon carbide composites, and low-activation steels. It could possibly be used for U.S./Japanese collaboration in the Jupiter Program. The first test is scheduled to be completed by September 1998. In this report, we present the functional requirements for the vehicle and a preliminary design that satisfies these requirements.

- 11.5 STATUS OF ATR-A1 IRRADIATION EXPERIMENT ON VANADIUM ALLOYS AND LOW-ACTIVATION STEELS — H. Tsai, R. V. Strain, I. Gomes, and D. L. Smith (Argonne National Laboratory), L. R. Greenwood (Pacific Northwest National Laboratory), and H. Matsui (Tohoku University, Japan) 303

The ATR-A1 irradiation experiment in the Advanced Test Reactor (ATR) was a collaborative U.S./Japan effort to study the effects of neutron damage on vanadium alloys at low temperature. The experiment also contained a limited quantity of low-activation ferritic steel specimens from Japan as part of the collaboration agreement. Irradiation was completed in 1996 after attaining the target exposure of  $\approx 4.7$  dpa in vanadium. The irradiated capsule was disassembled in this reporting period, and all specimens and monitors were successfully retrieved.

11.6	SCHEDULE AND STATUS OF IRRADIATION EXPERIMENTS — A. F. Rowcliffe, M. L. Grossbeck, and J. P. Robertson (Oak Ridge National Laboratory)	327
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The current status of reactor irradiation experiments is presented in tables summarizing the experimental objectives, conditions, and schedule.