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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 structure, has been completed at Wah Chang (formerly Teledyne Wah Chang) of Albany, Oregon (WCA). Joining of V-4Cr-4Ti alloy has been identified as the most critical fabrication issue for its use in the RD Program, and research into several joining methods for fabrication of the RD components, including resistance seam, friction, and electron beam welding, is continuing. Preliminary trials have been successful in the joining of V-alloy to itself by electron beam, resistance, and friction welding processes, and to Inconel 625 by friction welding. An effort to investigate the explosive bonding of V-4Cr-4Ti alloy to Inconel 625 has also been initiated, and results have been encouraging. In addition, preliminary tests have been completed to evaluate the susceptibility of V-4Cr-4Ti alloy to stress corrosion cracking in DIII-D cooling water, and the effects of exposure to DIII-D bakeout conditions on the tensile and fracture behavior of V-4Cr-4Ti alloy.
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- 1.5 MODE I AND MIXED MODE I/III CRACK INITIATION AND PROPAGATION BEHAVIOOF V-4Cr-4Ti ALLOY AT 25°C — H-X (Huaxin) Li, R. J. Kurtz and R. H. Jones (Pacific Northwest National Laboratory)

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The mode I and mixed-mode I/III fracture behavior of the production-scale heat (#832665) of V-4Cr-4Ti has been investigated at 25°C using compact tension (CT) specimens for a mode I crack and modified CT specimens for a mixed-mode I/III crack. The mode III to mode I load ratio was 0.47. Test specimens were vacuum annealed at 1000°C for 1 h after final machining. Both mode I and mixed-mode I/III specimens were fatigue cracked prior to J-integral testing. It was noticed that the mixed-mode I/III crack angle decreased from an initial 25 degrees to approximately 23 degrees due to crack plane rotation during fatigue cracking. No crack plane rotation occurred in the mode I specimen. The crack initiation and propagation behavior was evaluated by generating J-R curves. Due to the high ductility of this alloy and the limited specimen thickness (6.35 mm), plane strain requirements were not met so valid critical J-integral values were not obtained. However, it was found that the crack initiation and propagation behavior was significantly different between the mode I and the mixed-mode I/III specimens. In the mode I specimen crack initiation did not occur, only extensive crack tip blunting due to plastic deformation. During J-integral testing the mixed-mode crack rotated to an increased crack angle (in contrast to fatigue precracking) by crack blunting. When the crack initiated, the crack angle was about 30 degrees. After crack initiation the crack plane remained at 30 degrees until the test was completed. Mixed-mode crack initiation was difficult, but propagation was easy. The fracture surface of the mixed-mode specimen was characterized by microvoid coalescence.

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

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Oxidation studies were conducted on 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. The oxidation rates, calculated from parabolic kinetic measurements of thermogravimetric testing and confirmed by microscopic analyses of cross sections of exposed specimens, were 5, 17, and $27\ \mu\text{m}$ per year after exposure at 300 , 400 , and 500°C , respectively. Uniaxial tensile tests were conducted at room temperature and at 500°C on preoxidized specimens of the alloy to examine the effects of oxidation and oxygen migration on tensile strength and ductility. Correlations were developed between tensile strength and ductility of the oxidized alloy and microstructural characteristics such as oxide thickness, depth of hardened layer, depth of intergranular fracture zone, and tranverse crack length.

- 1.8 TENSILE PROPERTIES OF V-Cr-Ti ALLOYS AFTER EXPOSURE IN HELIUM AND LOW-PARTIAL-PRESSURE OXYGEN ENVIRONMENTS — K. Natesan and W. K. Soppet (Argonne National Laboratory) 40

A test program is in progress to evaluate the effect of oxygen at low pO_2 on the tensile properties of V-(4-5)wt.% Cr-(4-5)wt.% Ti alloys. Some of the tensile specimens were precharged with oxygen at low pO_2 at 500°C and reannealed in vacuum at 500°C prior to tensile testing. In another series of tests, specimens were exposed for 250-275 h at 500°C in environments with various pO_2 levels and subsequently tensile tested at room temperature. The preliminary results indicate that both approaches are appropriate for evaluating the effect of oxygen uptake on the tensile properties of the alloys. The data showed that in the relatively short-time tests conducted thus far, the maximum engineering stress slightly increased after oxygen exposure but the uniform and total elongation values exhibited significant decrease after exposure in oxygen-containing environments. The data for a specimen exposed to a helium environment were similar to those obtained in low pO_2 environments.

- 1.9 MEASUREMENT OF HYDROGEN SOLUBILITY AND DESORPTION RATE IN V-4Cr-4Ti AND LIQUID LITHIUM-CALCIUM ALLOYS — J.-H. Park, R. Erck, E.-T. Park, S. Crossley, and F. Deleglise (Argonne National Laboratory) 45

Hydrogen solubility in V-4Cr-4Ti and liquid lithium-calcium was measured at a hydrogen pressure of 9.09×10^{-4} torr at temperatures between 250 and 700°C. Hydrogen solubility in V-4Cr-4Ti and liquid lithium decreased with temperature. The measured desorption rate of hydrogen in V-4Cr-4Ti is a thermally activated process; the activation energy is 0.067 eV. Oxygen-charged V-4Cr-4Ti specimens were also investigated to determine the effect of oxygen impurity on hydrogen solubility and desorption in the alloy. Oxygen in V-4Cr-4Ti increases hydrogen solubility and desorption kinetics. To determine the effect of a calcium oxide insulator coating on V-4Cr-4Ti, hydrogen solubility in lithium-calcium alloys that contained 0-8.0 percent calcium was also measured. The distribution ratio R of hydrogen between liquid lithium or lithium-calcium and V-4Cr-4Ti increased as temperature decreased ($R \approx 10$ and 100 at 70 and 250°C , respectively). However at $<267^\circ\text{C}$, solubility data could not be obtained by this method because of the slow kinetics of hydrogen permeation through the vanadium alloy.

- 1.10 EVALUATING ELECTRICALLY INSULATING FILMS DEPOSITED ON V-4%Cr-4%Ti BY REACTIVE CVD — J.-H. Park and W. D. Cho (Argonne National Laboratory) 52

Previous CaO coatings on V-4%Cr-4%Ti exhibited high-ohmic insulator behavior even though a small amount of vanadium from the alloy was incorporated in the coating. However, when the vanadium concentration in the coating is >15 wt.%, the coating becomes conductive. When the vanadium concentration is high in localized areas, a calcium vanadate phase that exhibits semiconductor behavior can form. To explore this situation, CaO and Ca-V-O coatings were produced on vanadium alloys by chemical vapor deposition (CVD) and by a metallic-vapor process to investigate the electrical resistance of the coatings. Initially, the vanadium alloy specimens were either charged with oxygen in argon that contained trace levels of oxygen, or oxidized for 1.5-3 h in a 1% CO-CO₂ gas mixture or in air to form vanadium oxide at 625-650°C. Most of the specimens were exposed to calcium vapor at 800-850°C. Initial and final weights were obtained to monitor each step, and surveillance samples were removed for examination by optical and scanning electron microscopy and electron-energy-dispersive and X-ray diffraction analysis; the electrical resistivity was also measured. We found that Ca-V-O films exhibited insulator behavior when the ratio of calcium concentration to vanadium concentration R in the film was >0.9 , and semiconductor or conductor behavior for $R < 0.8$. However, in some cases,

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dependent amorphization behavior of SiC irradiated with 0.56 MeV silicon ions at 1×10^{-3} dpa/s and with fission neutrons irradiated at 1×10^{-6} dpa/s irradiated to 15 dpa in the temperature range of $\sim 340 \pm 10$ K.

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- 2.6 EFFECTS OF NEUTRON IRRADIATION ON THE STRENGTH OF CONTINUOUS FIBER REINFORCED SiC/SiC COMPOSITES — G. E. Youngblood, C. H. Henager, Jr., and R. H. Jones (Pacific Northwest National Laboratory) 117

Flexural strength data as a function of irradiation temperature and dose for a SiC_f/SiC composite made with Nicalon-CG fiber suggest three major degradation mechanisms. Based on an analysis of tensile strength and microstructural data for irradiated Nicalon-CG and Hi-Nicalon fibers, it is anticipated that these degradation mechanisms will be alleviated in Hi-Nicalon reinforced composites.

- 2.7 THE MONOTONIC AND FATIGUE BEHAVIOR OF CFCCs AT AMBIENT TEMPERATURE AND 1000°C - N. Miriyala, P. K. Liaw, and C. J. McHargue (University of Tennessee), and L. L. Snead (Oak Ridge National Laboratory) 124

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Uniaxial tension creep response is reported for an oxide dispersion strengthened (ODS) steel, Fe-13.5Cr-2W-0.5Ti-0.25 Y₂O₃ (in weight percent) manufactured using the mechanical alloying process. Acceptable creep response is obtained at 900°C.

- 3.2 FRACTURE TOUGHNESS OF THE IEA HEAT OF F82H FERRITIC/MARTENSITIC STAINLESS STEEL AS A FUNCTION OF LOADING MODE — Huaxin Li, D. S. Gelles (Pacific Northwest Laboratories), J. P. Hirth (Washington State University-Pullman), and R. H. Jones (Pacific Northwest Laboratories) 142

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- 3.3 SUMMARY OF THE IEA WORKSHOP/WORKING GROUP MEETING ON FERRITIC/MARTENSITIC STEELS FOR FUSION — R. L. Klueh (Oak Ridge National Laboratory) 147

An International Energy Agency (IEA) Working Group on Ferritic/Martensitic Steels for Fusion Applications, consisting of researchers from Japan, the European Union, the United States, and Switzerland, met at the headquarters of the Joint European Torus (JET), Culham, United Kingdom, 24-25 October 1996. At the meeting, preliminary data generated on the large heats of steel purchased for the IEA program and on other heats of steels were presented and discussed. The second purpose of the meeting was to continue planning and coordinating the collaborative test program in progress on reduced-activation ferritic/martensitic steels. The next meeting will be held in conjunction with the International Conference on Fusion Reactor Materials (ICFRM-8) in Sendai, Japan, 23-31 October 1997.

- 3.4 FURTHER CHARPY IMPACT TEST RESULTS OF LOW ACTIVATION FERRITIC ALLOYS IRRADIATED AT 430°C TO 67 dpa — L. E. Schubert, M. L. Hamilton, and D. S. Gelles (Pacific Northwest National Laboratory) 151

Miniature CVN specimens of four ferritic alloys, GA3X, F82H, GA4X, and HT9, have been impact tested following irradiation at 430°C to 67 dpa. Comparison of the results with those of the previously tested lower dose irradiation condition indicates that the GA3X and F82H alloys, two primary candidate low activation alloys, exhibit virtually identical behavior following irradiation at 430°C to ~67 dpa and at 370°C to ~15 dpa. Very little shift is observed in either DBTT or USE relative to the unirradiated condition. The shifts in DBTT and USE observed in both GA4X and HT9 were smaller after irradiation at 430°C to ~67 dpa than after irradiation at 370°C at ~15 dpa.

- 3.5 THE EFFECT OF FUSION-RELEVANT HELIUM LEVELS ON THE MECHANICAL PROPERTIES OF ISOTOPICALLY TAILORED FERRITIC ALLOYS — G. L. Hankin (IPTME), M. L. Hamilton, D. S. Gelles (Pacific Northwest National Laboratory), and M. B. Toloczko (Washington State University) 156

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- 3.6 IRRADIATION CREEP OF VARIOUS FERRITIC ALLOYS IRRADIATED AT $\sim 400^\circ\text{C}$ IN THE PFR AND FFTF REACTORS — M. B. Toloczko and F. A. Garner (Pacific Northwest National Laboratory), and C. R. Eiholzer (Westinghouse Hanford Company) 162

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The unirradiated tensile properties of wrought GlidCop AL25 (ITER grade zero, IG0), solutionized and aged CuCrZr, and cold-worked and aged and solutionized and aged Hycon 3HP™ CuNiBe have been measured over the temperature range of 20-500°C at strain rates between $4 \times 10^{-4} \text{ s}^{-1}$ and 0.06 s^{-1} . The measured room temperature electrical conductivity ranged from 64 to 90% IACS for the different alloys. All of the alloys were relatively insensitive to strain rate at room temperature, but the strain rate sensitivity of GlidCop AL25 increased significantly with increasing temperature. The CuNiBe alloys exhibited the best combination of high strength and high conductivity at room temperature. The strength of CuNiBe decreased slowly with increasing temperature. However, the ductility of CuNiBe decreased rapidly with increasing temperature due to localized deformation near grain boundaries, making these alloy heats unsuitable for typical structural applications above 300°C. The strength and uniform elongation of GlidCop AL25 decreased significantly with increasing temperature at a strain rate of $1 \times 10^{-3} \text{ s}^{-1}$, whereas the total elongation was independent of test temperature. The strength and ductility of CuCrZr decreased slowly with increasing temperature.

- 4.2 FRACTURE TOUGHNESS OF COPPER-BASE ALLOYS FOR ITER APPLICATIONS: A PRELIMINARY REPORT — D. J. Alexander, S. J. Zinkle, and A. F. Rowcliffe (Oak Ridge National Laboratory) 175

Oxide-dispersion strengthened copper alloys and a precipitation-hardened copper-nickel-beryllium alloy showed a significant reduction in toughness at elevated temperatures (250°C). This decrease in toughness was much larger than would be expected from the relatively modest changes in the tensile properties over the same temperature range. However, a copper-chromium-zirconium alloy strengthened by precipitation showed only a small decrease in toughness at the higher temperatures. The embrittled alloys showed a transition in fracture mode, from transgranular microvoid coalescence at room temperature to intergranular with localized ductility at high temperatures. The Cu-Cr-Zr alloy maintained the ductile microvoid coalescence failure mode at all test temperatures.

- 4.3 RECENT RESULTS ON THE NEUTRON IRRADIATION OF ITER CANDIDATE COPPER ALLOYS IRRADIATED IN DR-3 AT 250°C TO 0.3 dpa — D. J. Edwards (PNNL), B. N. Singh, P. Toft, and M. Eldrup (Risø National Laboratory) 183

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The main effect of the bonding thermal cycle heat treatment was a slight decrease in strength of CuCrZr and CuNiBe alloys. The strength of CuAl-25, on the other hand, remained almost unaltered. The post irradiation tests at 250°C showed a severe loss of ductility in the

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Irradiation effects in materials depend in a complex way on the form of the as-produced primary damage state and its spatial and temporal evolution. Thus, while collision cascades produce defects on a time scale of tens of picoseconds, diffusion occurs over much longer time scales, of the order of seconds, and microstructure evolution over even longer time scales. In this report we present work aimed at describing damage production and evolution in metals across all the relevant time and length scales. We discuss results of molecular dynamics simulations of displacement cascades in Fe and V. We show that interstitial clusters are produced in cascades above 5 keV, but not vacancy clusters. Next, we discuss the development of a kinetic Monte Carlo model that enables calculations of damage evolution over much longer time scales (1000's of s) than the picosecond lifetime of the cascade. We demonstrate the applicability of the method by presenting predictions on the fraction of freely migrating defects in a-Fe during irradiation at 600 K [1].

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- 11.0 IRRADIATION FACILITIES, TEST MATRICES, AND EXPERIMENTAL METHODS 247

- 11.1 STATUS OF DOE/JAERI COLLABORATIVE PROGRAM PHASE II AND PHASE III CAPSULES — J. P. Robertson, K. E. Lenox (Oak Ridge National Laboratory), I. Ioka and E. Wakai (Japan Atomic Energy Research Institute) 249

Significant progress has been made during the last year in the post-irradiation examinations (PIE) of the specimens from nine DOE ORNL/JAERI collaborative capsules and in the design and fabrication of four additional capsules. JP21, JP22, CTR-62, and CTR-63 were disassembled, JP20 tensile specimens were tested, and a variety of specimens from the RB-60J-1, 200J-1, 330J-1, and 400J-1 capsules were tested. Fabrication of RB-11J and 12J was completed and progress made in the matrix finalization and design of RB-10J and JP25.

- 11.2 STATUS OF ATR-A1 IRRADIATION EXPERIMENT ON VANADIUM ALLOYS AND LOW-ACTIVATION STEELS — H. Tsai, R. V. Strain, I. Gomes, H. Chung, and D. L. Smith (Argonne National Laboratory), and H. Matsui (Tohoku University) 253

The ATR-A1 irradiation experiment in the Advanced Test Reactor (ATR) was a collaborative U.S./Japan effort to study at low temperature the effects of neutron damage on vanadium alloys. The experiment also contained a limited quantity of low-activation ferritic steel specimens from Japan as part of the collaboration agreement. The irradiation was completed on May 5, 1996, as planned, after achieving an estimated neutron damage of 4.7 dpa in vanadium. The capsule has since been kept in the ATR water canal for the required radioactivity cool-down. Planning is underway for disassembly of the capsule and test specimen retrieval.

- 11.3 PROGRESS REPORT ON THE DESIGN OF A VARYING TEMPERATURE IRRADIATION EXPERIMENT FOR OPERATION IN HFIR — A. L. Qualls (Oak Ridge National Laboratory), and T. Muroga (National Institute for Fusion Science) 255

A varying temperature irradiation experiment is being performed under the framework of the Japan-U.S. Program of Irradiation Tests for Fusion Research (JUPITER) to study the effects of temperature variation on the microstructure and mechanical properties of candidate fusion reactor structural materials. An irradiation capsule has been designed for operation in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) that will allow four sets of metallurgical test specimens to be irradiated to exposure levels ranging from 5 to 10 dpa. Two sets of specimens will be irradiated at constant temperatures of 500 and 350°C. Matching specimen sets will be irradiated to similar exposure levels, with 10% of the exposure to occur at reduced temperatures of 300 and 200°C.

- 11.4 DISASSEMBLY OF THE FUSION-1 CAPSULE AFTER IRRADIATION IN THE BOR-60 REACTOR — H. Tsai (Argonne National Laboratory), V. A. Kazakov and V. P. Chakin (Research Institute of Atomic Reactors), and A. F. Rowcliffe (Oak Ridge National Laboratory) 263

A U.S./Russia (RF) collaborative irradiation experiment, Fusion-1, was completed in June 1996 after reaching a peak exposure of ≈ 17 dpa in the BOR-60 fast reactor at the Research Institute of Atomic Reactors (RIAR) in Russia. The specimens were vanadium alloys, mainly of recent heats from both countries. In this reporting period, the capsule was disassembled at the RIAR hot cells and all test specimens were successfully retrieved. For the disassembly, an innovative method of using a heated diffusion oil to melt and separate the lithium bond from the test specimens was adopted. This method proved highly successful.

- 11.5 SCHEDULE AND STATUS OF IRRADIATION EXPERIMENTS — A. F. Rowcliffe and M. L. Grossbeck (Oak Ridge National Laboratory) 265

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