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- 1.2 A MASTER CURVE-MECHANISM BASED APPROACH TO MODELING THE EFFECTS OF CONSTRAINT, LOADING RATE AND IRRADIATION ON THE TOUGHNESS-TEMPERATURE BEHAVIOR OF A V-4Cr-4Ti ALLOY — G. R. Odette, E. Donahue, G. E. Lucas, and J. W. Scheckherd (University of California, Santa Barbara) 11
- The influence of loading rate and constraint on the effective fracture toughness as a function of temperature [$K_e(T)$] of the fusion program heat of V-4Cr-4Ti was measured using subsized, three point bend specimens. The constitutive behavior was characterized as a function of temperature and strain rate using small tensile specimens. Data in the literature on this alloy was also analyzed to determine the effect of irradiation on $K_e(T)$ and the energy temperature (E-T) curves measured in subsized Charpy V-notch tests. It was found that V-4Cr-4Ti undergoes "normal" stress-controlled cleavage fracture below a temperature marking a sharp ductile-to-brittle transition. The transition temperature is increased by higher loading rates, irradiation hardening and triaxial constraint. Shifts in a reference transition temperature due to higher loading rates and irradiation can be reasonably predicted by a simple equivalent yield stress model. These results also suggest that size and geometry effects, which mediate constraint, can be modeled by combining local critical stressed area σ^*/A^* fracture criteria with finite element method simulations of crack tip stress fields. The fundamental understanding reflected in these models will be needed to develop $K_e(T)$ curves for a range of loading rates, irradiation conditions, structural size scales and geometries relying (in large part) on small specimen tests. Indeed, it may be possible to develop a master $K_e(T)$ curve-shift method to account for these variables. Such reliable and flexible failure assessment methods are critical to the design and safe operation of defect tolerant vanadium structures.
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(HT1) or 1050°C for two hours (HT2). Specimens given the HT1 treatment were annealed after final machining, whereas the HT2 specimens received the 1050°C anneal at Teledyne Wah Chang prior to final machining. Following machining HT2 specimens were vacuum annealed at 180°C for two hours to remove hydrogen. Specimens treated using HT1 had a partially recrystallized microstructure and those treated using HT2 had a fully recrystallized microstructure. The fracture toughness at 25°C was determined by J-integral tests and at -196°C by ASTM 399 type tests. Toughness values obtained at -196°C were converted to J-integral values for comparison to the 25°C data. The 25°C fracture toughness was very high with none of the specimens giving valid results per ASTM criteria. Specimens fractured by microvoid coalescence. The fracture toughness at -196°C was much lower than that at 25°C and the fracture surface showed predominantly cleavage features. The present results show a transition from ductile to brittle behavior with decreasing test temperature which is not observed from one-third scale Charpy impact tests. The fracture toughness at -196°C was still quite high, however, at about 75 kJ/m².

Delaminations in planes normal to the thickness direction were seen at both test temperatures. Fracture surfaces inside the delaminations exhibited nearly 100% cleavage facets. The cause of the brittle delaminations was not determined, but will be a subject for further investigation.

1.4 EFFECT OF TIME AND TEMPERATURE ON GRAIN SIZE OF V AND V-Cr-Ti ALLOYS — K. Natesan and D. L. Rink (Argonne National Laboratory)

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Grain growth studies were conducted to evaluate the effect of time and temperature on the grain size of pure V, V-4 wt.%Cr-4 wt.%Ti, and V-5 wt.%Cr-5 wt.%Ti alloys. The temperatures used in the study were 500, 650, 800, and 1000°C, and exposure times ranged between 100 and ≈5000 h. All three materials exhibited negligible grain growth at 500, 650, and 800°C, even after ≈5000 h. At 1000°C, pure V showed substantial grain growth after only 100 h, and V-4Cr-4Ti showed growth after 2000 h, while V-5Cr-5Ti showed no grain growth after exposure for up to 2000 h.

1.5 EFFECTS OF STRAIN RATE, TEST TEMPERATURE AND TEST ENVIRONMENT ON TENSILE PROPERTIES OF VANADIUM ALLOYS — A. N. Gubbi, A. F. Rowcliffe, W. S. Eatherly, and L. T. Gibson (Oak Ridge National Laboratory)

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Tensile testing was carried out on SS-3 tensile specimens punched from 0.762-mm-thick sheets of the large heat of V-4Cr-4Ti and small heats of V-3Cr-3Ti and V-6Cr-6Ti. The tensile specimens were annealed at 1000° for 2 h to obtain a fully recrystallized, fine grain microstructure with a grain size in the range of 10-19 μm. Room temperature tests at strain rates ranging from 10⁻³ to 5 × 10⁻¹/s were carried out in air; elevated temperature testing up to 700°C was conducted in a vacuum better than 1 × 10⁻⁵ torr (<10⁻³ Pa). To study the effect of atomic hydrogen on ductility, tensile tests were conducted at room temperature in an ultra high vacuum chamber (UHV) with a hydrogen leak system.

Tensile properties of V-3Cr-3Ti, V-4Cr-4Ti, and V-6Cr-6Ti were measured at room temperature and 100–700°C at a strain rate of 1.1 × 10⁻³/s. The ultimate tensile strength of all the alloys exhibited a minima at 300°C, whereas the 0.2% yield strength was relatively independent of temperature between 400° and 700°C. The total and uniform elongations were relatively insensitive to variation in test temperature above 400°C. All the alloys exhibited good ductility (e.g., uniform elongation >15%) and a large amount of work hardening ability. A yield point

was typically obtained at all test temperatures. Serrations, indicative of dynamic strain aging, were observed in the stress-strain curves of all the alloys at test temperatures above 300°C.

V-6Cr-6Ti is the strongest of the three alloys with the highest values of 0.2% yield strength (YS) and the ultimate tensile strength (UTS), and V-3Cr-3Ti is the weakest showing the lowest values at all strain rates; V-4Cr-4Ti possesses intermediate strength. Both YS and UTS showed a similar trend of incremental increase with strain rate for the three alloys. All three alloys exhibited almost no change in uniform and total elongations up to a strain rate of $10^{-1}/s$ followed by a decrease with further increase in strain rate. The room temperature tensile behavior of V-4Cr-4Ti was unaffected by the introduction of a significant partial pressure of atomic hydrogen into the testing environment.

- 1.6 DEVELOPMENT OF LASER WELDING TECHNIQUES FOR VANADIUM ALLOYS
— R. V. Strain, K. H. Leong, and D. L. Smith (Argonne National Laboratory) 53

The development of techniques for joining vanadium alloys, and possibly vanadium, to steel will be required for the construction of fusion devices. The primary objective of this program is to develop laser welding techniques for vanadium alloys, and to evaluate the performance of weldments.

- 1.7 IMPACT PROPERTIES AND HARDENING BEHAVIOR OF LASER AND ELECTRON-BEAM WELDS OF V-4Cr-4Ti — H. M. Chung, R. V. Strain, H.-C. Tsai, J.-H. Park, and D. L. Smith (Argonne National Laboratory) 55

We are conducting a program to develop an optimal laser welding procedure that can be applied to large-scale fusion-reactor structural components to be fabricated from vanadium-base alloys. Results of initial investigation of mechanical properties and hardening behavior of laser and electron-beam (EB) welds of the production-scale heat of V-4Cr-4Ti (500-kg Heat #832665) in as-welded and postwelding heat-treated (PWHT) conditions are presented in this paper. The laser weld was produced in air using a 6-kW continuous CO₂ laser at a welding speed of ≈ 45 mm/s. Microhardness of the laser welds was somewhat higher than that of the base metal, which was annealed at a nominal temperature of $\approx 1050^\circ\text{C}$ for 2 h in the factory. In spite of the moderate hardening, ductile-brittle transition temperatures (DBTTs) of the initial laser ($\approx 80^\circ\text{C}$) and EB ($\approx 30^\circ\text{C}$) welds were significantly higher than that of the base metal ($\approx 170^\circ\text{C}$). However, excellent impact properties, with DBTT $< 80^\circ\text{C}$ and similar to those of the base metal, could be restored in both the laser and EB welds by postwelding annealing at 1000°C for 1 h in vacuum.

- 1.8 MICROSTRUCTURAL CHARACTERISTICS AND MECHANISM OF TOUGHNESS IMPROVEMENT OF LASER AND ELECTRON-BEAM WELDS OF V-4Cr-4Ti FOLLOWING POSTWELDING HEAT-TREATMENT — H. M. Chung, J.-H. Park, J. Gazda, and D. L. Smith (Argonne National Laboratory) 59

We are conducting a program to develop an optimal laser welding procedure for large-scale fusion-reactor structural components to be fabricated from vanadium-base alloys. Microstructural characteristics were investigated by optical microscopy, X-ray diffraction, transmission electron microscopy, and chemical analysis to provide an understanding of the mechanism of the drastic improvement of impact toughness of laser and electron-beam (EB) welds of V-4Cr-4Ti following postwelding annealing at 1000°C . Transmission electron microscopy (TEM) revealed that annealed weld zones were characterized by extensive networks of fine

V(C,O,N) precipitates, which appear to clean away O, C, and N from grain matrices. This process is accompanied by simultaneous annealing-out of the dense dislocations present in the weld fusion zone. It seems possible to produce high-quality welds under practical conditions by controlling and adjusting the cooling rate of the weld zone by some innovative method to maximize the precipitation of V(C,O,N).

- 1.9 EVALUATION OF FLOW PROPERTIES IN THE WELDMENTS OF VANADIUM ALLOYS USING A NOVEL INDENTATION TECHNIQUE — A. N. Gubbi, A. F. Rowcliffe, W. S. Eatherly, and L. T. Gibson (Oak Ridge National Laboratory)

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Automated Ball Indentation (ABI) testing, was successfully employed to determine the flow properties of the fusion zone, heat affected zone (HAZ), and base metal of the gas tungsten arc (GTA) and electron beam (EB) welds of the V-4Cr-4Ti (large heat no. 832665) and the V-5Cr-5Ti (heat 832394) alloys. ABI test results showed a clear distinction among the properties of the fusion zone, HAZ, and base metal in both GTA and EB welds of the two alloys. GTA and EB welds of both V-4Cr-4Ti and V-5Cr-5Ti alloys show strengthening of both the fusion zone and the HAZ (compared to base metal) with the fusion zone having higher strength than the HAZ. These data correlate well with the Brinell hardness. On the other hand, GTA welds of both alloys, after a post-weld heat treatment of 950°C for 2 h, show a recovery of the properties to base metal values with V-5Cr-5Ti showing a higher degree of recovery compared to V-4Cr-4Ti. These measurements correlate with the reported recovery of the Charpy impact properties.^{1,2}

- 1.10 PROPERTIES OF V-(8-9)Cr-(5-6)Ti ALLOYS IRRADIATED IN THE DYNAMIC HELIUM CHARGING EXPERIMENT — H. M. Chung, L. Nowicki, and D. L. Smith (Argonne National Laboratory)

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In the Dynamic Helium Charging Experiment (DHCE), helium was produced uniformly in vanadium alloy specimens by the decay of tritium during irradiation to 18-31 dpa at 425-600°C in lithium-filled capsules in the Fast Flux Test Facility. This report presents results of postirradiation tests of tensile properties and density change in V-8Cr-6Ti and V-9Cr-5Ti. Compared to tensile properties of the alloys irradiated in the non-DHCE (helium generation negligible), the effect of helium on tensile strength and ductility of V-8Cr-6Ti and V-9Cr-5Ti was insignificant after irradiation and testing at 420, 500, and 600°C. Both alloys retained a total elongation of >11% at these temperatures. Density change was <0.48% for both alloys.

- 1.11 TENSILE PROPERTIES OF V-(4-15)Cr-5Ti ALLOYS IRRADIATED AT 400°C IN THE HFIR — H. M. Chung, L. Nowicki, and D. L. Smith (Argonne National Laboratory)

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V-(4-15)Cr-5Ti alloys were irradiated in a helium environment to $\approx 400^\circ\text{C}$ in the High Flux Isotope Reactor (HFIR). This report presents results of postirradiation tests of tensile properties of V-4Cr-4Ti, V-8Cr-6Ti, V-10Cr-5Ti, and V-15Cr-5Ti. Despite concerns on the effects of transmutation of vanadium to Cr and impurity pickup from the helium environment, all of the alloys exhibited ductile tensile behavior. However, the alloys exhibited ductilities somewhat lower than those of the specimens irradiated to a similar dose and at a similar temperature in an Li environment in fast reactors. Uniform plastic strain in the V-Cr-(4-5)Ti alloy decreased monotonically with increasing Cr content.

- 1.12 EFFECTS OF IRRADIATION AT LOW TEMPERATURE ON V-4Cr-4Ti —
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- 1.13 GRAIN BOUNDARY MIGRATION INDUCED SEGREGATION IN V-Cr-Ti ALLOY —
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- 1.15 STUDY OF IN-REACTOR CREEP OF VANADIUM ALLOY IN THE HFIR RB-12J
EXPERIMENT — R. V. Strain, C. F. Konicek, and H. Tsai (Argonne National
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- 1.16 OXIDATION KINETICS AND MICROSTRUCTURE OF V-(4-5) WT.%Cr-(4-5) WT.%Ti
ALLOYS EXPOSED TO AIR AT 300-650°C — K. Natesan (Argonne National
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- A systematic study was conducted to determine the effects of time and temperature of air exposure on the oxidation behavior and microstructure of V-4r-4Ti (44) and V-5Cr-5Ti (55) alloys. All samples were from 1-mm-thick cold-rolled sheets, and each was annealed in vacuum at 1050°C for 1 h prior to high-temperature exposure. Different samples from each alloy were heated in ambient air at 500°C for times

ranging from 24 to ≈ 2000 h, and in thermogravimetric analysis (TGA) apparatus at 300 to 650°C. Models describing the oxidation kinetics, the oxide type and its thickness, alloy grain size, and the depth of oxygen diffusion in the substrate alloy were determined for the two alloys and compared. The results showed that the oxide layers that formed on the surfaces of both alloys in air in the temperature range of 300-650°C are protective, and that the 55 alloy is slightly more oxidation-resistant than the 44 alloy.

- 1.17 CaO INSULATOR COATINGS ON A VANADIUM-BASE ALLOY IN LIQUID 2 at.% CALCIUM-LITHIUM — J.-H. Park and T. F. Kassner (Argonne National Laboratory)

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The electrical resistance of CaO coatings produced on V-4%Cr-4%Ti and V-15%Cr-5%Ti by exposure of the alloy (round bottom samples 6-in. long by 0.25-in. dia) to liquid lithium that contained 2 at.% dissolved calcium was measured as a function of time at temperatures between 300-464°C. The solute element, calcium in liquid lithium, reacted with the alloy substrate at these temperatures for 17 h to produce a calcium coating $\approx 7-8$ μm thick. The calcium-coated vanadium alloy was oxidized to form a CaO coating. Resistance of the coating layer on V-15Cr-5Ti, measured in-situ in liquid lithium that contained 2 at.% calcium, was 1.0×10^{10} $\Omega\text{-cm}^2$ at 300°C and 400 h, and 0.9×10^{10} $\Omega\text{-cm}^2$ at 464°C and 300 h. Thermal cycling between 300 and 464°C changed the resistance of the coating layer, which followed insulator behavior. Examination of the specimen after cooling to room temperature revealed no cracks in the CaO coating. The coatings were evaluated by optical microscopy, scanning electron microscopy (SEM), electron dispersive spectroscopy (EDS), and X-ray analysis. Adhesion between CaO and vanadium alloys was enhanced as exposure time increased.

2.0 SILICON CARBIDE COMPOSITE MATERIALS

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- 2.1 REVISED ACTIVATION ESTIMATES FOR SILICON CARBIDE — H. L. Heinisch (Pacific Northwest National Laboratory), E. T. Cheng (TSI Research), and F. M. Mann (Westinghouse Hanford Company)

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Recent progress in nuclear data development for fusion energy systems includes a reevaluation of neutron activation cross sections for silicon and aluminum. Activation calculations using the newly compiled Fusion Evaluated Nuclear Data Library result in calculated levels of ^{26}Al in irradiated silicon that are about an order of magnitude lower than the earlier calculated values. Thus, according to the latest internationally accepted nuclear data, SiC is much more attractive as a low activation material, even in first wall applications.

- 2.2 TIME-DEPENDENT BRIDGING AND LIFE PREDICTION OF SiC/SiC IN A HYPOTHETICAL FUSION ENVIRONMENT — C. H. Henager, Jr., C. A. Lewinshon, C. F. Windisch, Jr., and R. H. Jones (Pacific Northwest National Laboratories)

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Growth of subcritical cracks in SiC/SiC composites of CG-Nicalon fibers with a ~ 1 μm C-interphase has been measured on a related Basic Energy Sciences program using environments of purified argon and mixtures of argon and oxygen at 1073K and 1373K. Companion thermo-gravimetric (TGA) testing measured mass loss in identical environments. The TGA mass loss was from C-interphase oxidation to CO and CO₂, which was undetectable in argon and linear with oxygen concentration in argon-oxygen mixtures, and was converted into an interphase linear recession rate. Crack growth in pure argon indicated that fiber creep was causing time-dependent crack bridging to occur, while crack growth in argon-oxygen mixtures indicated that time-dependent C-interphase recession was also

causing time-dependent bridging with different kinetics. A model of time-dependent bridging was used to compute crack growth rates in argon and in argon-oxygen mixtures and gave an estimate of usable life of about 230 days at 1073K in a HE + 1.01 Pa O₂ (10 ppm) environment).

- 2.3 THE CYCLIC FATIGUE BEHAVIOR OF A NICALON/SiC COMPOSITE — N. Miriyala, P. K. Liaw, and C. J. McHargue (University of Tennessee) and L. L. Snead (Oak Ridge National Laboratory) 130

Cyclic fatigue tests were performed at ambient temperature on a Nicalon/SiC composite to study the effects of fabric orientation on the mechanical behavior. Four-point bend specimens were loaded either parallel or normal to the braided fabric plies. The maximum stress chosen during the fatigue tests were 60, 70, and 80% of the monotonic strengths, respectively, in both orientations. Specimen failure did not occur in any case even after one million loading cycles. However, it was observed that much of the decrease in the composite modulus occurred in the first few (<10 cycles), and the fabric orientation did not significantly effect the effective modulus or midspan deflection trends.

- 2.4 NEUTRON IRRADIATION EFFECTS ON HIGH NICALON SILICON CARBIDE FIBERS — M. C. Osborne, D. Steiner (Rensselaer Polytechnic Institute), and L. L. Snead (Oak Ridge National Laboratory) 136

The effects of neutron irradiation on the mechanical properties and microstructure of SiC and SiC-based fibers is a current focal point for the development of radiation damage resistant SiC/SiC composites. This report discusses the radiation effects on the Nippon Carbon Hi-NicalonTM fiber system as also discusses an erratum in earlier results published by the authors on this material. The radiation matrix currently under study is also summarized.

- 2.5 SPECIMEN SIZE EFFECT CONSIDERATIONS FOR IRRADIATION STUDIES OF SiC/SiC — G. E. Youngblood, C. H. Henegar, Jr., and R. H. Jones (Pacific Northwest National Laboratories) 140

For characterization of the irradiation performance of SiC/SiC, limited available irradiation volume generally dictates that tests be conducted on a small number of relatively small specimens. Flexure testing of two groups of bars with different sizes cut from the same SiC/SiC plate suggested the following lower limits for flexure specimen number and size: six simples at a minimum for each condition and a minimum bar size of 30 × 6.0 × 2.0 mm³.

- 2.6 TECHNIQUE FOR MEASURING IRRADIATION CREEP IN POLYCRYSTALLINE SiC FIBERS — G. E. Youngblood, M. L. Hamilton, and R. H. Jones (Pacific Northwest National Laboratory) 146

A bend stress relaxation (BSR) test has been designed to examine irradiation enhanced creep in polycrystalline SiC fibers being considered for fiber reinforcement in SiC/SiC composite. Thermal creep results on Nicalon-CG and Hi-Nicalon were shown to be consistent with previously published data with Hi-Nicalon showing about a 100°C improvement in creep resistance. Preliminary data were also obtained on Nicalon-S that demonstrated that its creep is greater than that of Hi-Nicalon.

- 2.7 PROGRESS IN THE DEVELOPMENT OF A SiC_f/SiC CREEP TEST —
M. L. Hamilton, C. A. Lewinsohn, R. H. Jones, G. E. Youngblood, and F. A. Garner
(Pacific Northwest National Laboratory) and S. L. Hecht (Westinghouse Hanford Company) 152

An effort is now underway to design an experiment that will allow the irradiation creep behavior of SiC_f/SiC composites to be quantified. Numerous difficulties must be overcome to achieve this goal, including determining an appropriate specimen geometry that will fit in the irradiation volumes available and developing a fabrication procedure for such a specimen. A specimen design has been selected, and development of fabrication methods is proceeding. Thermal and stress analyses are being performed to evaluate the viability of the specimen and to assist with determining the design parameters. A possible alternate type of creep test is also being considered. Progress in each of these areas is described in this report.

- 3.0 FERRITIC-MARTENSITIC STEELS 159

- 3.1 TENSILE AND CHARPY IMPACT PROPERTIES OF IRRADIATED REDUCED-ACTIVATION FERRITIC STEELS — R. L. Klueh and D. J. Alexander (Oak Ridge National Laboratory) 161

Tensile tests were conducted on eight reduced-activation Cr-W steels after irradiation to 15-17 and 26-29 dpa, and Charpy impact tests were conducted on the steels irradiated to 26-29 dpa. Irradiation was in the Fast Flux Test Facility at 365°C on steels containing 2.25-12% C4, varying amounts of W, V, and Ta, and 0.1%C. Previously, tensile specimens were irradiated to 6-8 dpa and Charpy specimens to 6-8, 15-17, and 20-24 dpa. Tensile and Charpy specimens were also thermally aged to 20000 h at 365°C. Thermal aging had little effect on the tensile behavior or the ductile-brittle transition temperature (DBTT), but several steels showed a slight increase in the upper-shelf energy (USE). After ≈7 dpa, the strength of the steels increased and then remained relatively unchanged through 26-29 dpa (i.e., the strength saturated with fluence). Postirradiation Charpy impact tests after 26-29 dpa showed that the loss of impact toughness, as measured by an increase in DBTT and a decrease in the USE, remained relatively unchanged from the values after 20-24 dpa, which had been relatively unchanged from the earlier irradiations. As before, the two 9Cr steels were the most irradiation resistant.

- 3.2 CHARPY IMPACT TEST RESULTS OF FOUR LOW ACTIVATION FERRITIC ALLOYS IRRADIATED AT 370°C TO 15 DPA — L. E. Schubert, M. L. Hamilton, and D. S. Gelles (Pacific Northwest National Laboratory) 171

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- 3.3 **FRACTOGRAPHIC EXAMINATION OF REDUCED ACTIVATION FERRITIC/MARTENSITIC STEEL CHARPY SPECIMENS IRRADIATED TO 30 DPA AT 370°C** — D. S. Gelles and M. L. Hamilton (Pacific Northwest National Laboratory) and L. E. Schubert (University of Missouri, Rolla) 177
- Fractographic examinations are reported for a series of reduced activation ferritic/martensitic steel Charpy impact specimens tested following irradiation to 30 dpa at 370°C in FFTF. One-third size specimens of six low activation steels developed for potential application as structural materials in fusion reactors were examined. A shift in brittle fracture appearance from cleavage to grain boundary failure was noted with increasing manganese content. The results are interpreted in light of transmutation induced composition changes in a fusion environment. -
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- 3.5 **HEAT-TO-HEAT VARIABILITY OF IRRADIATION CREEP AND SWELLING OF HT9 IRRADIATED TO HIGH NEUTRON FLUENCE AT 400-600°C** — M. B. Toloczko and F. A. Garner (Pacific Northwest National Laboratory) 189
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- 3.7 **MECHANICAL PROPERTIES AND MICROSTRUCTURE OF F-82H WELDED JOINTS USING CO₂ LASER BEAM** — N. Yamanouchi and K. Shiba (Japan Atomic Energy Research Institute) 195
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- 3.8 THE CONSEQUENCES OF HELIUM PRODUCTION ON MICROSTRUCTURAL DEVELOPMENT IN ISOTOPICALLY TAILORED FERRITIC ALLOYS — D. S. Gelles (Pacific Northwest National Laboratory)

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A series of alloys have been made adding various isotopes of nickel in order to vary the production of helium during irradiation by a two step nuclear reaction in a mixed spectrum reactor. The alloys use a base composition of Fe-12Cr with an addition of 1.5% nickel, either in the form of ^{60}Ni which produces no helium, ^{59}Ni which produces helium at a rate of about 10 appm He/dpa, or natural nickel ($^{\text{Nat}}\text{Ni}$) which provides an intermediate level of helium due to delayed development of ^{59}Ni . Specimens were irradiated in the HFIR at Oak Ridge, TN, to ≈ 7 dpa at 300 and 400°C. Microstructural examinations indicated that nickel additions promote precipitation in all alloys, but the effect appears to be much stronger at 400°C than at 300°C. There is sufficient dose by 7 dpa (and with 2 appm He) to initiate void swelling in ferritic/martensitic alloys. Little difference was found between response from ^{59}Ni and $^{\text{Nat}}\text{Ni}$. Also, helium bubble development for high helium generation conditions appeared to be very different at 300 and 400°C. At 300°C, it appeared that high densities of bubbles formed whereas at 400°C, bubbles could not be identified, possibly because of the complexity of the microstructure, but more likely because helium accumulated at precipitate interfaces.

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- 4.1 TENSILE AND ELECTRICAL PROPERTIES OF UNIRRADIATED AND IRRADIATED HYCON 3HP™ CuNiBe — S. J. Zinkle and W. S. Eatherly (Oak Ridge National Laboratory)

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The unirradiated tensile properties of two different heats of Hycon 3HP™ CuNiBe have been measured over the temperature range of 20-500°C for longitudinal and long transverse orientations. The room temperature electrical conductivity has also been measured for both heats. Both heats exhibited a very good combination of strength and conductivity at room temperature. The strength remained relatively high at all test temperatures, with a yield strength of 420-520 MPa at 500°C. However, low levels of ductility (<5% uniform elongation) were observed at test temperatures above 200-250°C, due to flow localization adjacent to grain boundaries. Fission neutron irradiation to a dose of ~ 0.7 dpa at temperatures between 100 and 240°C produced a slight increase in strength and a significant decrease in ductility. The measured tensile elongation increased with increasing irradiation temperature, with a uniform elongation of $\sim 3.3\%$ observed at 240°C. The electrical conductivity decreased slightly following irradiation, due to the presence of defect clusters and Ni, Zn, Co transmutation products. The data indicate that CuNiBe alloys have irradiated tensile and electrical properties comparable or superior to CuCrZr and oxide dispersion strengthened copper at temperatures <250°C, and may be suitable for certain fusion energy structural applications.

- 4.2 FRACTURE TOUGHNESS OF OXIDE-DISPERSION STRENGTHENED COPPER — D. J. Alexander (Oak Ridge National Laboratory)

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The fracture toughness of an oxide-dispersion strengthened copper alloy AL-15 has been examined at room temperature and 250°C, in air and in vacuum ($<10^{-6}$ torr). Increasing test temperature causes a significant decrease in the fracture toughness of this material, in either air or vacuum environments. In addition, specimens oriented in the T-L orientation (crack growth parallel to the extrusion direction)

show significantly lower toughness than those in the L-T orientation (crack growth perpendicular to the extrusion direction).

- 4.3 EFFECTS OF BONDING BAKEOUT THERMAL CYCLES ON PRE-AND POSTIRRADIATION MICROSTRUCTURES, PHYSICAL, AND MECHANICAL PROPERTIES OF COPPER ALLOYS — B. N. Singh, M. Eldrup, and P. Toft (Risø National Laboratory), and D. J. Edwards (Pacific Northwest National Laboratory) 221

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- 4.4 INFLUENCE OF NICKEL AND BERYLLIUM CONTENT ON SWELLING BEHAVIOR OF COPPER IRRADIATED WITH FAST NEUTRONS — B. N. Singh (Risø National Laboratory), F. A. Garner and D. J. Edwards (Pacific Northwest National Laboratory), and J. H. Evans (University of London) 222

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- 5.1 TEMPERATURE DEPENDENCE OF THE DEFORMATION BEHAVIOR OF 316 STAINLESS STEEL AFTER LOW TEMPERATURE NEUTRON IRRADIATION — J. E. Pawel-Robertson (Oak Ridge National Laboratory), I. Ioka (Japan Atomic Energy Research Institute), A. F. Rowcliffe, M. L. Grossbeck (Oak Ridge National Laboratory), and S. Jitsukawa (Japan Atomic Energy Research Institute) 225

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- 5.2 TENSILE PROPERTIES OF A TITANIUM MODIFIED AUSTENITIC STAINLESS STEEL AND THE WELD JOINTS AFTER NEUTRON IRRADIATION — K. Shiba, I. Ioka, S. Jitsukawa, S. Hamada, A. Hishinuma (Japan Atomic Energy Institute), and J. Pawel-Robertson (Oak Ridge National Laboratory (ORNL)) 239

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- 5.3 IRRADIATION CREEP AND SWELLING OF VARIOUS AUSTENITIC ALLOYS IRRADIATED IN PFR AND FFTF — F. A. Garner and M. B. Toloczko (Pacific Northwest National Laboratory), B. Munro and S. Adaway (AEA Technology), and J. Standring (UKAEA, retired) 251
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- 6.2 SUMMARY OF THE IRRADIATION HISTORY OF THE TRIST-ER1 CAPSULE — A. L. Qualls, W. S. Eatherly, D. W. Heatherly, M. T. Hurst, D. G. Raby, R. G. Sitterson, L. L. Snead, K. R. Thoms, R. L. Wallace, and S. J. Zinkle (Oak Ridge National Laboratory) 267
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- 6.3 SUMMARY OF ROUND ROBIN MEASUREMENTS OF RADIATION INDUCED CONDUCTIVITY IN WESGO AL995 ALUMINA — S. J. Zinkle (Oak Ridge National Laboratory) 272

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- 6.4 OPTICAL ABSORPTION OF NEUTRON-IRRADIATED SILICA FIBERS — D. W. Cooke, E. H. Farnum, and B. L. Bennett (Los Alamos National Laboratory) 275

Induced-loss spectra of silica-based optical fibers exposed to high (10^{23} n-m⁻³) and low (10^{21} n-m⁻²) fluences of neutrons at the Los Alamos Spallation Radiation Effects Facility (LASREF) have been measured. Two types of fibers consisting of a pure fuses silica core with fluorine-doped (~4 mole %) cladding were obtained from Fiberguide Industries and used in the as-received condition. Anhydroguide™ and superguide™ fibers contained less than 1 ppm, and 600 to 800 ppm of OH, respectively. The data suggest that presently available silica fibers can be used in plasma diagnostics, but the choice and suitability depends upon the spectral region of interest. Low-OH content fibers can be used for diagnostic purposes in the interval ~800 to 1400 nm if the exposure is to high-fluence neutrons. For low-fluence neutron exposures, the low-OH content fibers are best suited for use in the interval ~800 to 2000 nm, and the high-OH content fibers are the choice for the interval ~400 to 800 nm.

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- 8.1 AN INTEGRATED APPROACH TO ASSESSING THE FRACTURE SAFE MARGINS OF FUSION REACTOR STRUCTURES — G. R. Odette (University of California, Santa Barbara) 283

Design and operation of fusion reactor structures will require an appropriate data base closely coupled to a reliable failure analysis method to safely manage irradiation embrittlement. However, ongoing irradiation programs will not provide the information on embrittlement necessary to accomplish these objectives. A new engineering approach is proposed based on the concept of a master toughness-temperature curve indexed on an absolute temperature scale using shifts to account for variables such as size scales, crack geometry and loading rates as well as embrittlement. While providing a simple practical engineering expedient, the proposed method can also be greatly enhanced by fundamental mechanism based models of fracture and embrittlement. Indeed, such understanding is required for the effective use of small specimen test methods, which is an integral element in developing the necessary data base.

- 8.2 DEFECT INTERACTIONS WITHIN A GROUP OF SUBCASCADES — H. L. Heinisch
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- 9.1 NEUTRON DOSIMETRY AND DAMAGE CALCULATIONS FOR THE HFIR-JP-23
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- 9.2 A REVALUATION OF HELIUM/DPA RATIOS FOR FAST REACTOR AND
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- 11.1 STATUS OF ATR-A1 IRRADIATION EXPERIMENT OF VANADIUM ALLOYS AND
LOW-ACTIVATION STEELS — H. Tsai, R. V. Strain, I. Gomes, and D. L. Smith
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The ATR-A1 irradiation experiment 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 started in the Advanced Test Reactor (ATR) on November 30, 1995, and ended as planned on May 5, 1996. Total exposure was 132.9 effective full power days (EFPDs) and estimated neutron damage in the vanadium was 4.7 dpa. The vehicle has been discharged from the ATR core and is scheduled to be disassembled in the next reporting period.

- 11.2 FEASIBILITY OF CONDUCTING A DYNAMIC HELIUM CHARGING EXPERIMENT FOR VANADIUM ALLOYS IN THE ADVANCED TEST REACTOR — H. Tsai, I. Gomes, R. V. Strain, and D. L. Smith (Argonne National Laboratory), and H. Matsui (Tohoku University)

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The feasibility of conducting a dynamic helium charging experiment (DHCE) for vanadium alloys in the water-cooled Advanced Test Reactor (ATR) is being investigated as part of the U.S./Monbuscho collaboration. Preliminary findings suggest that such an experiment is feasible, with certain constraints. Creating a suitable irradiation position in the ATR, designing an effective thermal neutron filter, incorporating thermocouples for limited specimen temperature monitoring, and handling of tritium during various phases of the assembly and reactor operating all appear to be feasible. An issue that would require special attention, however, is tritium permeation loss through the capsule wall at the higher design temperatures ($\geq 600^{\circ}\text{C}$). If permeation is excessive, the reduced amount of tritium entering the test specimens would limit the helium generation rates in them. At the lower design temperatures ($\approx 425^{\circ}\text{C}$), sodium, instead of lithium, may have to be used as the bond material to overcome the tritium solubility limitation.

- 11.3 SCHEDULE AND STATUS OF IRRADIATION EXPERIMENTS — A. F. Rowcliffe (ORNL)

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