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- 1.1 TENSILE AND IMPACT PROPERTIES OF GENERAL ATOMICS 832864 HEAT OF V-4Cr-4Ti ALLOY — H. Tsai, L. J. Nowicki, J. Gazda, M. C. Billone, and D. L. Smith (Argonne National Laboratory), W. R. Johnson and P. Trester (General Atomics) 3

A 1300-kg heat of V-4Cr-4Ti alloy was procured by General Atomics (GA) for the DIII-D radiative divertor program. To determine the mechanical properties of this alloy, tensile and Charpy tests were conducted on specimens prepared from pieces of 4.8-mm-thick as-rolled plates, a major product form for the DIII-D application. The tensile tests were conducted at three temperatures, 26, 280, and 380°C, the last two being the anticipated peak temperatures during DIII-D boronization and postvent bake-out, respectively. Results from these tests show that the tensile and impact properties of the 832864 heat are comparable to those of the other smaller V-(4-5)Cr-(4-5)Ti alloy heats previously developed by the U.S. Fusion Materials Program and that scale-up of vanadium alloy production can be successfully achieved as long as reasonable process control is implemented.

- 1.2 HIGH TEMPERATURE TENSILE PROPERTIES OF V-4Cr-4Ti — S. J. Zinkle, A. F. Rowcliffe and C. O. Stevens (Oak Ridge National Laboratory) 11

Tensile tests have been performed on V-4Cr-4Ti at 750 and 800°C in order to extend the data base beyond the current limit of 700°C. From comparison with previous measurements, the yield strength is nearly constant and tensile elongations decrease slightly with increasing temperature between 300 and 800°C. The ultimate strength exhibits an apparent maximum near 600°C (attributable to dynamic strain aging) but adequate strength is maintained up to 800°C. The reduction in area measured on tensile specimens remained high (~80%) for test temperatures up to 800°C, in contrast to previous reported results.

- 1.3 TENSILE PROPERTIES OF VANADIUM-BASE ALLOYS IRRADIATED IN THE FUSION-1 LOW-TEMPERATURE EXPERIMENT IN THE BOR-60 REACTOR — H. Tsai, J. Gazda, L. J. Nowicki, M. C. Billone, and D. L. Smith (Argonne National Laboratory) 15

The irradiation has been completed and the test specimens have been retrieved from the lithium-bonded capsule at the Research Institute of Atomic Reactors (RIAR) in Russia. During this reporting period, the Argonne National Laboratory tensile specimens were received from RIAR and initial testing and examination of these specimens at ANL has been completed. The results, corroborating previous findings, showed a significant loss of work hardening capability in the materials. There appears to be no significant difference in behavior among the various heats of vanadium-base alloys in the V-(4-5)Cr-(4-5)Ti composition range. The variations in the preirradiation annealing conditions also produced no notable differences.

- 1.4 ROOM-TEMPERATURE FRACTURE IN V-(4-5)Cr-(4-5)Ti TENSILE SPECIMENS IRRADIATED IN FUSION-1 BOR-60 EXPERIMENT — J. Gazda and M. Meshii (Northwestern University) and H. Tsai (Argonne National Laboratory) 20

Specimens of V-(4-5)Cr-(4-5) Ti alloys were irradiated to ≈ 18 dpa at 320°C in the Fusion-1 capsule inserted into the BOR-60 reactor. Tensile tests at 23°C indicated dramatic yield strength increase (300%), lack of work hardening, and minimal (<1%) total elongations. SEM analysis of fracture side surfaces were conducted to determine reduction in area and the mode of fracture. The reduction of area was

negligible. All but one specimen failed by a combination of ductile shear deformation and cleavage crack growth. Transgranular cleavage cracks were initiated by stress concentrations at the tips of the shear bands. In side-view observations, evidence was found of slip bands typically associated with dislocation channeling. No differences due to pre-irradiation heat treatment and heat-to-heat composition variations were detected. The only deviation from this behavior was found in V-4Cr-4Ti-B alloy, which failed in the grip portion by complete cleavage cracking.

- 1.5 EFFECTS OF IRRADIATION TO 4 DPA AT 390°C ON THE FRACTURE TOUGHNESS OF VANADIUM ALLOYS — E. E. Gruber, T. M. Galvin, and O. K. Chopra (Argonne National Laboratory) 28

Fracture toughness J-R curve tests were conducted at room temperature on disk-shaped compact-tension DC(T) specimens of three vanadium alloys having a nominal composition of V-4Cr-4Ti. The alloys in the nonirradiated condition showed high fracture toughness; J_{IC} could not be determined but is expected to be above 600 kJ/m². The alloys showed very poor fracture toughness after irradiation to 4 dpa at 390°C, e.g., J_{IC} values of ≈ 10 kJ/m² or lower.

- 1.6 EFFECT OF IRRADIATION TEMPERATURE AND STRAIN RATE ON THE MECHANICAL PROPERTIES OF V-4Cr-4Ti IRRADIATED TO LOW DOSES IN FISSION REACTORS — S. J. Zinkle, L. L. Snead, A. F. Rowcliffe, D. J. Alexander, and L. T. Gibson (Oak Ridge National Laboratory) 33

Tensile tests performed on irradiated V-(3-6%)Cr-(3-6%)Ti alloys indicate that pronounced hardening and loss of strain hardening capacity occurs for doses of 0.1-20 dpa at irradiation temperatures below $\sim 330^\circ\text{C}$. The amount of radiation hardening decreases rapidly for irradiation temperatures above 400°C , with a concomitant increase in strain hardening capacity. Low-dose (0.1-0.5 dpa) irradiation shifts the dynamic strain aging regime to higher temperatures and lower strain rates compared to unirradiated specimens. Very low fracture toughness values were observed in miniature disk compact specimens irradiated at $200\text{-}320^\circ\text{C}$ to $\sim 1.5\text{-}15$ dpa and tested at 200°C .

- 1.7 MICROSTRUCTURAL EXAMINATION OF V-(3-6%)Cr-(3-5%) Ti IRRADIATED IN THE ATR-A1 EXPERIMENT — D. S. Gelles (Pacific Northwest National Laboratory) 41

Microstructural examination results are reported for four heats of V-(3-6%)Cr-(3-5%)Ti irradiated in the ATR-A1 experiment to ~ 4 dpa at ~ 200 and 300°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 or composition.

- 1.8 IRRADIATION-INDUCED PRECIPITATION AND MECHANICAL PROPERTIES OF VANADIUM ALLOYS AT $<430^\circ\text{C}$ — H. M. Chung, J. Gazda, and D. L. Smith (Argonne National Laboratory) 49

Recent attention to V-base alloys has focused on the effect of low-temperature ($<430^\circ\text{C}$) irradiation on tensile and impact properties of V-4Cr-4Ti. In previous studies, dislocation channeling, which causes flow localization and severe loss of work-hardening capability, has been attributed to dense, irradiation-induced precipitation of very fine particles. However, efforts to identify the precipitates were

unsuccessful until now. In this study, analysis by transmission electron microscopy (TEM) was conducted on unalloyed V, V-5Ti, V-3Ti-1Si, and V-4Cr-4Ti specimens that were irradiated at $<430^{\circ}\text{C}$ in conventional and dynamic helium charging experiments. By means of dark-field imaging and selected-area-diffraction analysis, the characteristic precipitates were identified to be $(\text{V},\text{Ti}_{1-x})(\text{C},\text{O},\text{N})$. In V-3Ti-1Si, precipitation of $(\text{V},\text{Ti}_{1-x})(\text{C},\text{O},\text{N})$ was negligible at $<430^{\circ}\text{C}$, and as a result, dislocation channeling did not occur and work-hardening capability was high.

- 1.9 REACTIONS OF HYDROGEN WITH V-Cr-Ti ALLOYS — J. R. DiStefano, J. H. DeVan, L. D. Chitwood (Oak Ridge National Laboratory), and D. H. Röhrig (Projektleitung Kernfusion, Forschungszentrum Karlsruhe) 61

In the absence of increases in oxygen concentration, additions of up to 400 ppm hydrogen to V-4 Cr-4 did not result in significant embrittlement as determined by room temperature tensile tests. However, when hydrogen approached 700 ppm after exposure at 325°C , rapid embrittlement occurred. In this latter case, hydride formation is the presumed embrittlement cause. When oxygen was added during or prior to hydrogen exposure, synergistic effects led to significant embrittlement by 100 ppm oxygen.

- 1.10 TENSILE PROPERTIES OF V-Cr-Ti ALLOYS AFTER EXPOSURE IN HYDROGEN-CONTAINING ENVIRONMENTS — K. Natesan and W. K. Soppet (Argonne National Laboratory) 68

A systematic study has been initiated at Argonne National Laboratory to evaluate the performance of several V-Cr-Ti alloys after exposure to environments containing hydrogen at various partial pressures. The goal is to correlate the chemistry of the exposure environment with hydrogen uptake in the samples and its influence on the microstructure and tensile properties of the alloys. At present, the principal effort has focused on the V-4Cr-4Ti alloy of heat identified as BL-71; however other alloys (V-5Cr-5Ti alloy of heats BL-63, and T87, plus V-4Cr-4Ti alloy from General Atomics) are also being evaluated. Other variables of interest are the effect of initial grain size on hydrogen uptake and tensile properties, and the synergistic effects of oxygen and hydrogen on the tensile behavior of the alloys. Experiments conducted on specimens of various V-Cr-Ti alloys exposed to pH_2 levels of 0.01 and 3×10^{-6} torr showed negligible effect of H_2 on either maximum engineering stress or uniform and total elongation. However, uniform and total elongation decreased substantially when the alloys were exposed to 1.0 torr H_2 pressure. Preliminary data from sequential exposures of the materials to low- pO_2 and several low- pH_2 environments did not reveal an adverse effect on the maximum engineering stress or on uniform and total elongation. Further, tests in H_2 environments on specimens annealed at different temperatures showed that grain-size variation by a factor of ≈ 2 had little or no effect on tensile properties.

- 1.11 OXIDATION BEHAVIOR OF V-Cr-Ti ALLOYS IN LOW-PARTIAL-PRESSURE OXYGEN ENVIRONMENTS — K. Natesan and M. Uz (Argonne National Laboratory) 73

A test program is in progress at Argonne National Laboratory to evaluate the effect of pO_2 in the exposure environment on oxygen uptake, scaling kinetics, and scale microstructure in V-Cr-Ti alloys. The data indicate that the oxidation process follows parabolic kinetics in all of the environments used in the present study. From the weight change data, parabolic rate constants were evaluated as a function of temperature and exposure environment. The temperature dependence of the parabolic rate constants was described by an Arrhenium relationship. Activation energy for the oxidation process was fairly constant in the oxygen pressure range of

1×10^{-6} to 1×10^{-1} torr for both the alloys. The activation energy for oxidation in air was significantly lower than in low- pO_2 environments, and for oxidation in pure O_2 at 760 torr was much lower than in low- pO_2 environments. X-ray diffraction analysis of the specimens showed that VO_2 was the dominant phase in low- pO_2 environments, while V_2O_5 was dominant in air and in pure oxygen at 760 torr.

- 1.12 MICROSTRUCTURAL CHARACTERIZATION OF EXTERNAL AND INTERNAL OXIDE PRODUCTS ON V-4Cr-4Ti — B. A. Pint, P. M. Rice, L. D. Chitwood, J. H. DeVan, and J. R. DiStefano (Oak Ridge National Laboratory) 77

Air oxidation of V-4Cr-4Ti at 500°C at 1 atm resulted in the formation of a thin (100-150 nm) external vanadium nitride layer which was identified beneath a thicker (1.5 μ m) vanadium oxide scale. This nitride layer would only be detected by high-resolution, analytical electron microscopy techniques. Subsequent tests comparing room temperature tensile properties for exposure in laboratory air, dry air, and dry oxygen at 1 atm showed more embrittlement in air than in O_2 . Internal oxidation of coarse-grained V-4Cr-4Ti at low oxygen pressures at 500°C was followed by TEM examination. In a sample with a 1400 ppmw O addition, which is sufficient to reduce the ductility to near zero, there appeared to be an oxygen denuded zone (150-250 nm) near the grain boundaries with precipitates at the grain boundaries and uniform ultra-fine (<5 nm) oxygen particles in the matrix. In a similar O-loaded specimen that was subsequently annealed for 4 h at 950°C to restore ductility, large oxide particles were observed in the matrix and at the grain boundaries.

- 1.13 DEVELOPMENT OF ELECTRICALLY INSULATING CaO COATINGS — K. Natesan, C. B. Reed, M. Uz, and D. L. Rink (Argonne National Laboratory) 82

A systematic study has been initiated to develop electrically insulating CaO coatings by vapor phase transport and by in-situ formation in a liquid Li environment. Several experiments were conducted in vapor transport studies with variations in process temperature, time, specimen location, specimen surface preparation, and pretreatment. Several of the coatings obtained by this method exhibited Ca concentration in the range of 60-95 wt.% on the surface. However, coating thickness has not been very uniform among several samples exposed in the same run or even within the same sample. The coatings developed in these early tests degraded after 24 h exposure to Li at 500°C. Additional experiments are under way to develop better-adhering and more dense coatings by this method.

A program to develop in-situ CaO coating in Li has been initiated, and the first set of capsule tests at 800°C in three different Li-Ca mixtures will be completed in early July. Specimens included in the run are bare V-4Cr-4Ti alloy, specimens with a grit-blasted surface and O-precharged in 99.999% Ar, polished specimens precharged in a 99.999% Ar and 5000 ppm O_2-N_2 mixture, and prealuminized V-5Cr-5Ti alloy preoxidized in a 5000 ppm O_2-N_2 mixture. Additional experiments at lower temperatures are planned.

- 1.14 LASER-WELDED V-Cr-Ti ALLOYS: MICROSTRUCTURAL AND MECHANICAL PROPERTIES — K. Natesan, D. L. Smith, Z. Xu, and K. H. Leong (Argonne National Laboratory) 87

A systematic study has been in progress at Argonne National Laboratory to examine the use of YaG or CO_2 lasers to weld sheet materials of V-Cr-Ti alloys and to characterize the microstructural and mechanical properties of the laser-welded materials. In addition, several postwelding heat treatments are being applied to the welded samples to evaluate their benefits, if any, to the structure and properties of the weldments. Hardness measurements are made across the welded regions of

- different samples to evaluate differences in the characteristics of various weldments. Several weldments were used to fabricate specimens for four-point bend tests. Several additional weldments were made with a YAG laser; here, the emphasis was on determining the optimal weld parameters to achieve deep penetration in the welds. A preliminary assessment was then made of the weldments on the basis of microstructure, hardness profiles, and defects.
- 2.0 SILICON CARBIDE COMPOSITE MATERIALS** 91
- 2.1 THERMOPHYSICAL AND MECHANICAL PROPERTIES OF SiC/SiC COMPOSITES — S. J. Zinkle and L. L. Snead (Oak Ridge National Laboratory)** 93
- The key thermophysical and mechanical properties for SiC/SiC composites are summarized, including temperature-dependent tensile properties, elastic constants, thermal conductivity, thermal expansion, and specific heat. The effects of neutron irradiation on the thermal conductivity and dimensional stability (volumetric swelling, creep) of SiC is discussed. The estimated lower and upper temperatures limits for structural applications in high power density fusion applications are 400 and 1000°C due to thermal conductivity degradation and void swelling considerations, respectively. Further data are needed to more accurately determine these estimated temperature limits.
- 2.2 A REVIEW OF JOINING TECHNIQUES FOR SiC_F/SiC COMPOSITES FOR FIRST WALL APPLICATIONS — C. A. Lewinsohn and R. H. Jones (Pacific Northwest National Laboratory)** 101
- Many methods for joining monolithic and composite silicon carbide are available. Three techniques are candidates for use in fusion energy systems: in-situ displacement reactions, pre-ceramic polymer adhesives, and reaction bonding. None of the methods are currently developed enough to satisfy all of the criteria required, i.e., low temperature fabrication, high strength, and radiation stability.
- 2.3 THE HFIR 14J SiC/SiC COMPOSITE AND SiC FIBER COLLABORATION — G. E. Youngblood and R. H. Jones (Pacific Northwest National Laboratory), Akira Kohyama and Yutai Kato (Kyoto University), Akira Hasegawa (Tohoku University), Reinhard Scholz (European Joint Research Commission, and Lance Snead (Oak Ridge National Laboratory)** 115
- A short introduction with references establishes the current status of research and development of SiC_F/SiC composites for fusion energy systems with respect to several key issues. The SiC fiber and composite specimen types selected for the JUPITER 14J irradiation experiment are presented together with the rationale for their selection.
- 2.4 NEUTRON IRRADIATION INDUCED AMORPHIZATION OF SILICON CARBIDE — L. L. Snead and J. C. Hay (Oak Ridge National Laboratory)** 122
- This paper provides the first known observation of silicon carbide fully amorphized under neutron irradiation. Both high purity single crystal hcp and high purity, highly faulted (cubic) chemically vapor deposited (CVD) SiC were irradiated at approximately 60°C to a total fast neutron fluence of 2.6×10^{25} n/m². Amorphization was seen in both materials, as evidenced by TEM, electron diffraction, and x-ray diffraction techniques. Physical properties for the amorphized single crystal material are reported including large changes in density (-10.8%), elastic modulus as measured using a nanoindentation technique (-45%), hardness as measured by

nanindentation (-45%), and standard Vickers hardness (-24%). Similar property changes are observed for the amorphized CVD SiC. Using measured thermal conductivity data for the CVD SiC sample, the critical temperature for amorphization at this neutron dose and flux, above which amorphization is not possible, is estimated to be greater than 130°C.

3.0 FERRITIC/MARTENSITIC STEELS 133

3.1 THERMOPHYSICAL AND MECHANICAL PROPERTIES OF Fe-(8-9)%Cr REDUCED ACTIVATION STEELS — S. J. Zinkle, J. P. Robertson, and R. L. Klueh (Oak Ridge National Laboratory) 135

The key thermophysical and mechanical properties for 8-9%Cr reduced activation ferritic/ martensitic steels are summarized, including temperature-dependent tensile properties in the unirradiated and irradiated conditions, stress-rupture behavior, elastic constants, thermal conductivity, thermal expansion, specific heat, and ductile-to-brittle transition temperature. The estimated lower and upper temperatures limits for structural applications are 250 and 550°C due to radiation hardening/embrittlement and thermal creep considerations, respectively.

3.2 ANALYSIS OF STRESS-INDUCED BURGERS VECTOR ANISOTROPY IN PRESSURIZED TUBE SPECIMENS OF IRRADIATED FERRITIC-MARTENSITIC STEEL: JLF-1 — D. S. Gelles (Pacific Northwest National Laboratory) and T. Shibayama (University of Hokkaido, Japan) 144

A procedure for determining the Burgers vector anisotropy in irradiated ferritic steels allowing identification of all $a\langle 100 \rangle$ and all $\frac{a}{2}\langle 111 \rangle$ dislocations in a region of interest is applied to a pressurized tube specimen of JLF-1 irradiated at 430°C to 14.3×10^{22} n/cm² ($E > 0.1$ MeV) or 61 dpa. Analysis of micrographs indicates large anisotropy in Burgers vector populations develop during irradiation creep.

3.3 MECHANICAL PROPERTIES OF IRRADIATED 9Cr-3WVTa STEEL — R. L. Klueh and D. J. Alexander (Oak Ridge National Laboratory), and M. Rieth (Forschungszentrum Karlsruhe Institut für Materialforschung II) 150

An Fe-9Cr-2W-0.25V-0.07Ta-0.1C (9Cr-2WVTa) steel has excellent strength and impact toughness before and after irradiation in the Fast Flux Test Facility and the High Flux Reactor (HFR). The ductile-brittle transition temperature (DBTT) increased only 32°C after 28 dpa at 365°C in FFTF, compared to a shift of ≈60°C for a 9Cr-2WV steel—the same as the 9Cr-2WVTa steel but without tantalum. This difference occurred despite the two steels having similar tensile properties before and after irradiation. The 9Cr-2WVTa steel has a smaller prior-austenite grain size, but otherwise microstructures are similar before irradiation and show similar changes during irradiation. The irradiation behavior of the 9Cr-2WVTa steel differs from the 9Cr-2WV steel in two ways: (1) the shift in DBTT of the 9Cr-2WVTa steel irradiated in FFTF does not saturate with fluence by ≈28 dpa, whereas for the 9Cr-2WV steel and most similar steels, saturation occurs at <10 dpa, and (2) the shift in DBTT for 9Cr-2WVTa steel irradiated in FFTF and HFR increased with irradiation temperature, whereas it decreased for the 9Cr-2WV steel, as it does for most similar steels. The improved properties of the 9Cr-2WVTa steel and the differences with other steels were attributed to tantalum in solution.

- 3.4 MICROSTRUCTURAL ANALYSIS OF FERRITIC-MARTENSITIC STEELS IRRADIATED AT LOW TEMPERATURE IN HFIR — N. Hashimoto (Oak Ridge National Laboratory), E. Wakai (Japan Atomic Energy Research Institute), J. P. Robertson and A. F. Rowcliffe (Oak Ridge National Laboratory) 163

Disk specimens of ferritic-martensitic steel, HT9 and F82H, irradiated to damage levels of ~3 dpa at irradiation temperatures of either ~90°C or ~250°C have been investigated by using transmission electron microscopy. Before irradiation, tempered HT9 contained only $M_{23}C_6$ carbide. Irradiation at 90°C and 250°C induced a dislocation loop density of $1 \times 10^{22} \text{ m}^{-3}$ and $8 \times 10^{21} \text{ m}^{-3}$, respectively. In the HT9 irradiated at 250°C, a radiation-induced phase, tentatively identified as α' , was observed with a number density of less than $1 \times 10^{22} \text{ m}^{-3}$. Difference in the radiation-induced phase and the loop microstructure may be related to differences in the post-yield deformation behavior of the two steels.

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- 4.1 PROGRESS REPORT ON THE INFLUENCE OF TEST TEMPERATURE AND GRAIN BOUNDARY CHEMISTRY ON THE FRACTURE BEHAVIOR OF ITER COPPER ALLOYS — M. Li and J. F. Stubbins (University of Illinois) and D. J. Edwards (Pacific Northwest National Laboratory) 173

This collaborative study was initiated to determine mechanical properties at elevated temperatures of various copper alloys by University of Illinois and Pacific Northwestern National Lab with support of OMG Americas, Inc., and Brush Wellman, Inc. This report includes current experimental results on notch tensile tests and pre-cracked bend bar tests on these materials at room temperature, 200 and 300°C. The elevated temperature tests were performed in vacuum and indicate a decrease in fracture resistance with increasing temperature, as seen in previous investigations. While the causes for the decreases in fracture resistance are still not clear, the current results indicate that environmental effects are likely less important in the process than formerly assumed.

- 4.2 COMPARISON OF PROPERTIES AND MICROSTRUCTURES OF TRÉFIMÉTAUX AND HYCON 3HP™ AFTER NEUTRON IRRADIATION — D. J. EDWARDS (Pacific Northwest National Laboratory), B. N. Singh, P. Toft, and M. Eldrup (Risø National Laboratory) 183

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- 4.3 TENSILE AND ELECTRICAL PROPERTIES OF HIGH-STRENGTH HIGH-CONDUCTIVITY COPPER ALLOYS — S.J. Zinkle and W.S. Eatherly (Oak Ridge National Laboratory) 189

Electrical conductivity and tensile properties have been measured on an extruded and annealed CuCrNb dispersion strengthened copper alloy which has been developed for demanding aerospace high heat flux applications. The properties of this alloy are somewhat inferior to GlidCop dispersion strengthened copper and prime-aged CuCrZr over the temperature range of 20-500°C. However, if the property degradation in CuCrZr due to joining operations and the anisotropic properties of GlidCop in the short transverse direction are taken into consideration, CuCrNb may be a suitable alternative material for high heat flux structural applications in fusion energy devices. The electrical conductivity and tensile properties of CuCrZr that was solution annealed and then simultaneously aged and diffusion bonded are also summarized. A severe reduction in tensile elongation is

observed in the diffusion bonded joint, particularly if a thin copper shim is not placed in the diffusion bondline.

- 4.4 **ROUND ROBIN COMPARISON OF TENSILE RESULTS ON GlidCop Al25** —
D. J. Edwards (Pacific Northwest National Laboratory), S. J. Zinkle (Oak Ridge National Laboratory), S. A. Fabritsiev (DV Efremov Institute), and A. S. Pokrovsky (Research Institute of Atomic Reactors) 193

A round robin comparison of the tensile properties of GlidCop™ Al25 oxide dispersion strengthened copper was initiated between collaborating laboratories to evaluate the test and analysis procedures used in the irradiation experiments in SRIAR in Dimitrovgrad. The tests were conducted using the same tensile specimen geometry as used in previous irradiation experiments, with tests at each laboratory being conducted in air or vacuum at 25, 150, and 300°C at a strain rate of $3 \times 10^{-4} \text{ s}^{-1}$. The strength of the GlidCop™ Al25 decreased as the test temperature increased, with no observable effect of testing in air versus vacuum on the yield and ultimate strengths. The uniform elongation decreased by almost a factor of 3 when the test temperature was raised from room temperature to 300°C, but the total elongation remained roughly constant over the range of test temperatures. Any effect of testing in air on the ductility may have been masked by the scatter introduced into the results because each laboratory tested the specimens in a different grip setup. In light of this, the results of the round robin tests demonstrated that the test and analysis procedures produced essentially the same values for tensile yield and ultimate, but significant variability was present in both the uniform and total elongation measurements due to the gripping technique.

- 4.5 **PROGRESS REPORT ON THE BEHAVIOR AND MODELING OF COPPER ALLOY TO STAINLESS STEEL JOINTS FOR ITER FIRST WALL APPLICATIONS** — J. Min, J. Stubbins, J. Collins (University of Illinois), and A. F. Rowcliffe (Oak Ridge National Laboratory) 200

The stress states that lead to failure of joints between GlidCop™ CuAl25 and 316L SS were examined using finite element modeling techniques to explain experimental observations of behavior of those joints. The joints were formed by hot isostatic pressing (HIP) and bend bar specimens were fabricated with the joint inclined 45° to the major axis of the specimen. The lower surface of the bend bar was notched in order to help induce a precrack for subsequent loading in bending. The precrack was intended to localize a high stress concentration in close proximity to the interface so that its behavior could be examined without complicating factors from the bulk materials and the specimen configuration. Preparatory work to grow acceptable precracks caused the specimen to fail prematurely while the precrack was still progressing into the specimen toward the interface. This prompted the finite element model calculations to help understand the reasons for this behavior from examination of the stress states throughout the specimen. An additional benefit sought from the finite element modeling effort was to understand if the stress states in this non-conventional specimen were representative of those that might be experienced during operation in ITER.

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- 6.0 **INSULATING CERAMICS AND OPTICAL MATERIALS** 211

No contributions.

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The irradiation environment experienced by the in-vessel components of fusion reactors presents structural design challenges not envisioned in the development of existing structural design criteria such as the ASME Code or RCC-MR. From the standpoint of design criteria, the most significant issues stem from the irradiation-induced changes in materials properties. Specifically, the reduction of ductility, strain hardening capability, and fracture toughness with neutron irradiation. Recently, Draft 7 of the ITER structural design criteria (ISDC), which provide new rules for guarding against such problems, was released for trial use by the ITER designers. The new rules, which were derived from a simple model based on the concept of elastic follow up factor, provide primary and secondary stress limits as functions of uniform elongation and ductility. The implication of these rules on the allowable surface heat flux on typical first walls made of type 316 stainless steel and vanadium alloys are discussed.

- 10.2 ELASTIC-PLASTIC ANALYSIS OF THE SS-3 TENSILE SPECIMEN S. Majumdar (Argonne National Laboratory) 248

Tensile tests of most irradiated specimens of vanadium alloys are conducted using the miniature SS-3 specimen which is not ASTM approved. Detailed elastic-plastic finite element analysis of the specimen was conducted to show that, as long as the ultimate to yield strength ratio is less than or equal to 1.25 (which is satisfied by many irradiated materials), the stress-plastic strain curve obtained by using such a specimen is representative of the true material behavior.

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- 11.1 STATUS OF LITHIUM-FILLED SPECIMEN SUBCAPSULES FOR THE HFIR-MFE-RB-10J EXPERIMENT -- J. P. Robertson, M. Howell, and K. E. Lenox (Oak Ridge National Laboratory) 257

The HFIR-MFE-RB-10J experiment will be irradiated in a Removable Beryllium position in the HFIR for 10 reactor cycles, accumulating approximately 5 dpa in steel. The upper region of the capsule contains two lithium-filled subcapsules containing vanadium specimens. This report describes the techniques developed to achieve a satisfactory lithium fill with a specimen occupancy of 26% in each subcapsule.

- 11.2 SPECIMEN LOADING LIST FOR THE VARYING TEMPERATURE EXPERIMENT — A. L. Qualls and R. G. Sitterson (Oak Ridge National Laboratory) 260

The varying temperature experiment HFIR-RB-13J has been assembled and inserted in the reactor. Approximately 5300 specimens were cleaned, inspected, matched, and loaded into four specimen holders. A listing of each specimen loaded into the steady temperature holder, its position in the capsule, and the identification of the corresponding specimen loaded into the varying temperature holder is presented in this report.

- 11.3 STATUS OF THE IRRADIATION TEST VEHICLE FOR TESTING FUSION MATERIALS IN THE ADVANCED TEST REACTOR — H. Tsai, I. C. Gomes, and D. L. Smith (Argonne National Laboratory), A. J. Palmer, and F. W. Ingram (Lockheed Martin Idaho Technologies Company), and F. W. Wiffen (U.S. Department of Energy) 278

The design of the irradiation test vehicle (ITV) for the Advanced Test Reactor (ATR) has been completed. The main application for the ITV is irradiation testing of candidate fusion structural materials, including vanadium-base alloys, silicon

carbided composites, and low-activation steels. Construction of the vehicle is under way at the Lockheed Martin Idaho Technology Company (LMITCO). Dummy test trains are being built for system checkout and fine-tuning. Reactor insertion of the ITV with the dummy test trains is scheduled for fall 1998. Barring unexpected difficulties, the ITV will be available for experiments in early 1999.

11.4 SCHEDULE AND STATUS OF IRRADIATION EXPERIMENTS — A. F. Rowcliffe,
M. L. Grossbeck, and J. P. Robertson (Oak Ridge National Laboratory) 283

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