TABLE OF CONTENTS

1 FERRITIC/MARTENSITIC STEEL DEVELOPMENT

1.1 Development of Advanced Reduced Activation Ferritic/Martensitic Steels — L. Tan (Oak Ridge National Laboratory) **1**

New heats of CNAs have been fabricated, which showed superior yield and tensile strength at test temperatures up to 750°C. The superior strength is explained by their refined microstructure with significantly increased amount of MX-type nanoprecipitates. Charpy impact toughness of the CNAs was assessed in this work, which showed excellent upper shelf energies that are more than double that of Grade 91 steel and about 8–10 times that of oxide dispersion strengthened (ODS) steels. The ductile-brittle transition temperature (DBTT) of the CNAs is comparable or lower than Grade 91 steel. Characterization of the high-dose Fe^{2+} ion-irradiated samples containing VN nanoprecipitates exhibited evolution with irradiation depth, that is, with increasing irradiation dose. The growth of VN particles was found to be dependent on the irradiation direction.

1.2 New Bainitic Steel for Fusion Structural Applications — Y. Yamamoto (Oak Ridge National Laboratory) **4**

Four heats of 3Cr-3WV base bainitic steels were delivered to ORNL. Minor alloying additions of Mn, Si, and N did not affect the as-normalized and tempered microstructure, viewed in an optical microscope, but increased the micro-Vickers hardness. The bainitic transformation kinetics during continuous cooling were measured using a GleebleTM thermal mechanical simulator, which indicated that the transformation temperature was significantly lowered by the minor alloying additions. This indicates a wider window for the carbide-free acicular bainite ferrite formation in the modified steels compared to the base steels, which would play an important role in improving the creep properties. Creep-rupture property evaluation of these heats was initiated, and the tests of two base steels were completed.

1.3 Microstructural Characterization of a Fe+ Irradiated Fe-6%Cr Alloy — Y. Wu, P. Wells, G. R. Odette, T. Yamamoto (University of California Santa Barbara), S. Roberts and C. Hardie (University of Oxford), D. Bhattacharyya (Australian Nuclear Science and Technology Organization), M. Bachhav and E. A. Marquis (University of Michigan) **8**

Both ion-beam and neutron irradiations of candidate structural alloys produce large populations of interstitial dislocation loops that lead to hardening and other effects on mechanical properties. Here we report on the number density, size distribution and character of the loops produced by high energy Fe-ions in a Fe-6Cr binary alloy; and we further compare these results to the corresponding microstructures produced by neutron irradiation. While the two types of irradiations produce qualitatively similar features, there are possibly some notable differences. In particular, it appears that only Si segregates to the loops in the Fe-ion irradiations, while both Si and Cr are associated with the loops in the case of neutron irradiations.

TABLE OF CONTENTS

1.4 **On the Effects of Helium-DPA Interactions on Cavity Evolution in Tempered Martensitic Steels under Dual Ion-Beam Irradiation** — Takuya Yamamoto, Yuan Wu, G. Robert Odette (University of California Santa Barbara), Kiyohiro Yabuuchi, Sosuke Kondo, Akihiko Kimura (Kyoto University) **12**

Cavity evolution in normalized and tempered 8Cr martensitic steels (TMS) F82H under Fe^{3+} and He^{+} dual ion beam irradiation (DII) at 500°C was characterized for a number of new dpa and He conditions. The cavity evolution TMS alloy database at 500°C now includes 280 material-dpa-He-dpa rate DII conditions. In summary, key observations on swelling include: a) the f_v-dpa trend is controlled by the He/dpa; b) better accounting for local microstructure effects and run-to-run variations partly rationalized the observed large data f_v "scatter"; c) f_v is somewhat lower in the reference case and similar runs; d) f_v is lower in the Ta modified variant of F82H than in the IEA heat and f_v in the latter decreases with increasing cold work.

1.5 **Three Dimensional Fractography of HFIR-Irradiated F82Hs** – H. Sakasegawa, H.Tanigawa (Japan Atomic Energy Agency), and Y. Katoh (Oak Ridge National Laboratory) **18**

The 3D fractography was performed using a scanning electron microscope. Three pictures of tensile specimen rupture surfaces were taken from different tilt angles and 3D images were produced from these pictures using a software program. From the representative result shown here, it was revealed that the irradiated specimen of F82H-MOD3 heat irradiated to 2.7 dpa at 573 K showed a shear rupture mode, with a small area fraction of dimpled surface.

2 **ODS AND NANOCOMPOSITED ALLOY DEVELOPMENT**

2.1 **Development of ODS FeCrAl for Fusion Reactor Applications** – B. A. Pint, K. A. Unocic, S. Dryepondt and D. T. Hoelzer (Oak Ridge National Laboratory, USA) **22**

Characterization of the first four experimental ODS FeCrAl heats (based on Fe-12Cr-5Al) is nearing completion. Creep testing of the alloys containing Y2O3 + ZrO2 and Y2O3 + HfO2 at 800°C/100 MPa has shown exceptional lifetimes for these materials. Additional Pb-Li compatibility experiments were completed at 700°C including similar composition ODS FeCrAl alloys made for a nuclear energy project. All of the alloys showed low mass changes in these experiments. suggesting superior Pb-Li compatibility compared to wrought and ODS Fe-Cr compositions. A thin (~1 µm) reaction product of LiAlO2 was observed in all cases and additional characterization is in progress. The final phase of this project will examine two new alloys made with the same Fe-12Cr-5Al powder and several new alloys where Zr was added as an alloy addition rather than an oxide dispersion.

2.2 **Characterization of the Microstructure and Texture of NFA-1 for Two Deformation Processing Routes** — S. Pal, M.E. Alam, G. R. Odette (UCSB), J. Lewandowski (Case Western University) D. T. Hoelzer (ORNL) and S. A. Maloy (LANL) **29**

Our previous investigation showed that NFA-1 produced through ball milling and hot extrusion followed by hot cross rolling suffers from severe micro-cracking on planes parallel to the plate surface and normal to the thickness direction. The microcracking inhibits fabrication of components, such as thin wall tubing, by

TABLE OF CONTENTS

conventional processing routes. This work reveals that a strong {001}<110> brittle texture component develops in NFA-1 during thermo-mechanical processing. Dislocation pile-ups at low angle boundaries result in formation of cleavage micro-cracks on {001} planes. In contrast, tubes formed by hydrostatic extrusion develop a strong {110}<211> "J1-shear texture" component and low angle grain boundaries. Since preferred slip occurs on {110} planes in bcc Fe, deformation is readily accommodated by easy dislocations glide in the hydrostatic tube extrusion case. This is in contrast to {001} slip planes with pile ups that lead to stress concentrations and micro-crack nucleation in the extruded and cross rolled plate.

2.3 **Fabrication and Characterization of Fe–{100}YTO Bilayers** —T. Stan, Y. Wu, and G. R. Odette (University of California Santa Barbara), H. D. Zhou (University of Tennessee) **42**

Nanostructured Ferritic Alloys (NFAs) have a Fe-Cr matrix and are dispersion strengthened by < 5 nm Y-Ti-O nano-oxide (NO) phases. The interfaces between the Y-Ti-O NO, such as $Y_2Ti_2O_7$ (YTO), and the surrounding matrix provide many favorable characteristics pertinent to fusion environments, such as trapping He in fine-scale bubbles. As a supplement to current characterization efforts of the NOs themselves, surrogate bulk Fe-YTO interfaces have been fabricated by electron beam Fe deposition on {100} YTO substrates. Grains with the {110}Fe||{100}YTO and <111>Fe||<110>YTO orientation relationship (OR) matched the two-fold symmetry of the {100}YTO substrate. This OR is not observed in embedded NOs. These results will support the development of computational models to predict NFA behavior.

2.4 **Irradiation-Damage Behavior in Advanced Steels** — C. M. Parish, Y. Katoh, M. E. Bannister, K. A. Unocic, L. Tan, B.-K. Kim, D. T. Hoelzer (Oak Ridge National Laboratory), and S. J. Zinkle (University of Tennessee) **48**

Four iron-based alloys were investigated: a 9Cr NFA "9YWTV" containing Y-Ti-oxides, a 9Cr CNA containing TaC precipitates, and two Fe-12Cr-5Al ODS alloys containing mixed populations of Y-Al-oxides, Y-Zr-oxides and Y-Hf-oxides. In a D-T burning plasma environment, the 14 MeV neutrons produce copious helium transmutation and the materials ability to mitigate helium effects must be characterized for future fusion materials selection and development. Here, sequential He and Fe ion irradiation is combined with advanced post-irradiation characterization to compare the four materials' helium mitigation.

2.5 **Microstructural Summary of ODS Ferritic Alloys (14YW, 14YWT, 12YWT, MA957FR, PM2000) and RAFM Steels (F82H Mod.3-CW, Eurofer97) from JP27 In-Situ He Injection (ISHI) Experiment at 500°C** — H. J. Jung, D. J. Edwards, R. J. Kurtz (Pacific Northwest National Laboratory), G. R. Odette, Y. Wu, T. Yamamoto (University of California Santa Barbara) **55**

A TEM characterization of various ferritic-based alloys has been conducted to document the changes in microstructure under neutron irradiation with and without the impact of high levels of He injection. Microstructural features such as dislocation loops, network dislocations, He bubbles and voids were characterized in five ODS alloys, 14YW, 14YWT, 12YWT, MA957Fr, PM2000, and two RAFM steels, F82H mod.3+CW and Eurofer97. The total neutron dose was ~21.2 dpa and the total He level was ~1230 appm injected into an ~6 µm uniform region below the surface of the sample. The results indicated that He injection

TABLE OF CONTENTS

significantly impacts the microstructure evolution path, producing changes in the dislocation structure and cavity distribution that are closely related to the size, density and chemistry of the oxide particles, or the lack thereof in the case of the RAFM alloys.

2.6 Microstructure, Texturing, Microcracking and Delamination Behavior of NFA-1 —S. Pal, M. E. Alam and G. R. Odette (UCSB), D. Hoelzer (ORNL), S. Maloy (LANL) **66**

NFA-1 was processed by gas atomization of powders, mechanical alloying by powder ball milling, consolidation by hot extrusion, annealing and hot cross-rolling. NFA-1 has highly anisotropic microstructures and mechanical properties. View planes parallel to the plate side show pancake-shaped grains of ~ 0.5 μm average thickness and aspect ratios of 2-3. Larger and a few much larger grains are also seen**.** Extrusion and cross rolling induce a strong <110> fiber texture, accompanied by brittle {001}<110> cleavage system planes parallel to the plate surface. Sessile dislocations with a Burgers vector of <001> on {001} planes along low-angle subgrain boundaries form during the the initial stages of deformation. Further deformation results in dislocation pile-ups at the subgrain boundaries, creating stress concentrations and opening displacements that nucleate microcracks on the brittle cleavage system. Microcrack propagation is driven by residual stresses that develop during the deformation.

2.7 Crack Healing by Annealing in 14YWT Nanostructured Ferritic Alloy, FCRD NFA-1 — M.E. Alam, S. Pal, G. R. Odette (UCSB), D. T. Hoelzer (ORNL) and S. A. Maloy (LANL) **83**

FCRD NFA-1 is a new 14YWT nanostructured ferritic alloy (NFA) processed to form a 12.5 mm thick plate. Plane sections parallel to the broad plate surface have a nearly equiaxed, ultrafine grain structure. In contrast, the plane sections parallel to the narrow plate thickness side contain pancake shaped grains with a tri-modal size distribution, including some that are very large, and a population of microcracks running normal to the thickness direction. At 23°C tensile specimens loaded in the short thickness direction have flat, faceted fracture surfaces and no ductility due to cleavage initiation of the microcracks. The corresponding tensile ductility increases with test temperature due to a brittle to ductile transition near about ~150°C. Annealing at 1300°C/1h appears to heal the microcracks, resulting in ductile fracture for short thickness direction axis loading. Annealing also appears to alter the grains and reduces the room temperature (23°C) yield stress by ≈ 13%. Tensile tests at 600°C showed relatively isotropic properties in both conditions, and a smaller reduction in strength following annealing.

2.8 Friction Stir Welding of ODS Steels and Advanced Ferritic Structural Steels— Z. Feng, W. Tang, X. Yu, G. Chen, D. Hoelzer, and L. Tan (Oak Ridge National Laboratory) **91**

For this final reporting period of the project, the baseline FSW process conditions for joining ODS steels and RAFM steels have been successfully developed. A computational fluid dynamics (FCD) based model has been developed and applied to understand of temperature and material flow during FSW of these advanced structural materials. A unique high temperature digital image correlation (DIC) strain measurement technique enabled experimentally determining the local deformation behavior in different regions of FSW of RAFM steel in high

TABLE OF CONTENTS

temperature creep test. The DIC measurement revealed drastic increase in creep resistance (i.e. reduced creep rate) in the stir zone, and reduced creep resistance in the heat affected zone (HAZ), an area requiring further process optimization. Finally, the sizes of the oxides/nano-cluster features, the primary strengthening phase in ODS alloys, is highly dependent on the FSW process conditions. This shows the feasibility of using FSW to join advanced fusion structural materials with better than base metal properties.

3 CERAMIC COMPOSITE STRUCTURAL MATERIAL DEVELOPMENT

3.1 Development of SiC Joining Technologies for Fusion: Pre-Irradiation Experiment — T. Koyanagi, J.O. Kiggans, and Y. Katoh (Oak Ridge National laboratory), T. Hinoki (Kyoto University, Japan), M. Ferraris (Politecnico di Torino, Italy), C.H. Henager (Pacific Northwest National Laboratory), C.A. Lewinsohn (Ceramatec Inc.), S. Grasso (Queen Mary University of London) **99**

Various SiC joints were prepared using diffusion bonding with active titanium and molybdenum inserts, pressurized and pressureless transient eutectic-phase (TEP) sintering, pressurized and pressureless reaction-formed Ti-Si-C MAX-phase bonding, $CaO–Al_2O_3$ (CA) glass ceramics, and Al-Si-C-O braze. Torsional shear strength, fracture behavior, and microstructural of the bonding layer were evaluated in the unirradiated condition. These results are being used as the basis to understand neutron irradiation effects on mechanical properties and microstructures of the SiC joints.

3.2 Low Activation Joining of SiC/SiC Composites for Fusion Applications: Modeling Miniature Torsion Tests with Elastic and Elastic-Plastic Models — C.H. Henager, Jr., B.N. Nguyen, and R.J. Kurtz; (Pacific Northwest National Laboratory, Richland, WA, USA); M. Ferraris, (Politecnico di Torino, Torino, Italy); Y. Katoh, (Oak Ridge National Laboratory, Oak Ridge,TN, USA) **108**

The international fusion community designed miniature torsion specimens for joint testing and irradiation in test reactors with limited irradiation volumes since SiC and SiC-composites used in fission or fusion environments require joining methods for assembling systems. Torsion specimens fail out-of-plane when joints are strong and when elastic moduli are comparable to SiC, which causes difficulties in determining shear strengths for many joints or for comparing unirradiated and irradiated joints. A finite element damage model was developed to treat elastic joints such as SiC/Ti_3SiC_2+SiC and elastic-plastic joints such as SiC/epoxy and steel/epoxy. The model uses constitutive shear data and is validated using epoxy joint data. The elastic model indicates fracture is likely to occur within the joined pieces to cause out-of-plane failures for miniature torsion specimens when a certain modulus and strength ratio between the joint material and the joined material exists. Lower modulus epoxy joints always fail in plane and provide good model validation.

3.3 Progress in Characterization of Precipitates and Defect Structures in Mg^+ Ion Implanted Cubic Silicon Carbide — W. Jiang, J. Zhang, Z. Zhu, T. J. Roosendaal, S. Y. Hu, C. H. Henager, Jr., R. J. Kurtz (Pacific Northwest National Laboratory), and Y. Wang (Los Alamos National Laboratory) **119**

This report describes the progress of our current experimental effort on Mg^+ ion implanted 3C-SiC. Following our initial study [i] that suggests possible formation of

TABLE OF CONTENTS

Mg_2Si and MgC_2 precipitates as well as tetrahedral voids in $^{24}Mg^+$ ion implanted 3C-SiC, we have designed specific experiments to confirm the results and examine the inclusions and defects. Relatively low fluence (5.0×10^{15} $^{24}Mg^+/cm^2$) implantation in 3C-SiC was performed to reduce defect concentrations and isolate individual defect features for characterization. In addition, $^{25}Mg^+$ isotope was implanted in 3C-SiC to the same previously applied ion fluence (9.6×10^{16} ions/cm^2) for atom probe tomography (APT) study of precipitates. Each set of the samples was annealed at 1573 K for 2, 6 and 12 h, respectively. The depth profiles of the implanted Mg were measured using secondary ion mass spectrometry (SIMS) before and after the annealing steps. The samples are currently being analyzed using transmission electron microscopy (TEM) and APT.

4 HIGH HEAT FLUX MATERIALS AND COMPONENT TESTING

4.1 Recent Progress in the Fabrication and Characterization of Ductile-Phase-Toughened Tungsten Laminates for Plasma-Facing Materials — K. H. Cunningham, G. R. Odette, K. Fields, D. Gragg, T. Yamamoto, and F. W. Zok (University of California, Santa Barbara) **123**

A promising approach to increasing the fracture toughness of W-alloys is ductile-phase toughening (DPT). A ductile reinforcement in a brittle matrix increases toughness primarily by crack bridging. A W-Cu laminate was fabricated, and fracture resistance curves (R-curves) were calculated via precracked three-point bend testing. An analytical model of crack bridging was used to estimate the parameter space of useful toughening reinforcements for rolled W plate. Work began on extending the model framework to calculate the bridging stress-displacement function (bridging law) of a ductile reinforcement from test load-displacement data.

4.2 Recent Progress in the Development of Ductile-Phase Toughened Tungsten for Plasma-Facing Materials: W-Ni-Fe Composites — C. H. Henager, Jr., E. L. Stevens, R. J. Kurtz, T. J. Roosendaal, E. A. Nyberg, C. A. Lavender (Pacific Northwest National Laboratory), G. R. Odette, K. H. Cunningham, and F. W. Zok (University of California, Santa Barbara) **130**

A promising approach to increasing fracture toughness and decreasing the DBTT of a W-alloy is by ductile-phase toughening (DPT) . In this method, a ductile phase is included in a brittle matrix to prevent fracture propagation by crack bridging or crack deflection. Liquid-phase sintered W-Ni-Fe alloys consisting of nearly spherical W-particles embedded within a Ni-Fe-W ductile matrix are being manipulated by hot-rolling to create lamellar W/Fe-Ni-W composites with anisotropic fracture properties. The rolled W-Ni-Fe alloy becomes a lamellar alloy consisting of W lamellae separated by ductile-phase regions. The W-rich lamellae are strong but brittle, while the ductile-phase metallic regions have a thin, plate-like morphology to provide a ductile bridging region. This rolled material is oriented with the W-rich lamellae parallel to principal stresses so that surface cracking is normal to the ductile-phase bridging regions.

TABLE OF CONTENTS

4.3 **Process Improvements for Preparation of Nano-W Specimens for Mechanical Testing and Testing Under Plasma Environments**— C. Ren and Zak Fang, Department of Metallurgical Engineering, University of Utah, 135 S. 1460 E. Room 412, Salt Lake City, UT 84112 **140**

In the last reporting period, the problem of sample cracking during sintering was resolved by implementing the novel multi-step pelletizing and compaction process (MPC). The ongoing work in this report period is focused on improving the sintered density of nano-W specimens. To accomplish this objective, a systemic study on the effect of sintering temperature, sintering atmosphere, sintering time and reduction temperature on the densification behavior of nano-W materials has been performed.

4.4 **Fabrication of Functionally Graded Tungsten Steel Laminate** — L. M. Garrison (Oak Ridge National Laboratory) **146**

Tungsten foils in thicknesses 250, 100, and 25 µm and grade 92 steel foils in nominal thicknesses 250, 100, and 76 µm were obtained. The foils were alternately stacked within a stainless steel container and then hot rolled at 1000°C to approximately 80% reduction of the original height to induce bonding. The composite was analyzed with EDS to reveal the elemental composition at the tungsten-steel interfaces. Tungsten foils were electropolished to reveal the grains for EBSD analysis. Samples of the tungsten and steel foils were tensile tested. An initial test was done with select foil samples utilizing digital image correlation to monitor the deformation.

4.5 **Defect Evolution in Neutron-Irradiated Single Crystalline Tungsten** — X. Hu, T. Koyanagi, Y. Katoh (Oak Ridge National Laboratory), M. Fukuda (Tohoku University), B.D. Wirth, L.L. Snead (University of Tennessee, Knoxville) **154**

High purity <110> single crystalline tungsten samples were examined by positron annihilation spectroscopy following low temperature and low dose neutron irradiation and subsequent isochronal annealing. The results were used to determine microstructural and defect evolutions, thus, identifying the annealing mechanisms. The one-hour isochronal annealing was performed on two specimens neutron-irradiated to very low (0.006) and low (0.03) dpa at ~90°C. Following annealing at 400, 500, 650, 800, 1000, 1150, and 1300°C, *ex-situ* characterization of vacancy defects used positron lifetime spectroscopy and coincidence Doppler broadening (CDB). Vacancy cluster size distributions showed significant damage recovery around 1000°C, consistent with Stage III below 400°C and Stage V temperature around 1000°C. CDB results confirmed the trend of the vacancy defect evolution and S-W plots confirmed a single type of vacancy clusters. This microstructure information is consistent with the measured hardness changes.

4.6 **Neutron Irradiation Effects in Tungsten** — L.M. Garrison, P. Edmondson, N.A.P. Kiran Kumar, T. Colling, M. McAlister, L. Snead, T. S. Byun, D. Lewis (Oak Ridge National Laboratory), and M. Fukuda (Tohoku University, Japan) **161**

A total of 440 samples were irradiated in HFIR at temperatures from 70 to 900°C and fast neutron fluences of 0.01 to 20 $\times 10^{25}$ n/m^2 at E>0.1 MeV. Types of tungsten irradiated in this study were [110] single crystal tungsten, [100] single crystal tungsten, wrought tungsten foils, annealed tungsten foils, and tungsten-

TABLE OF CONTENTS

copper laminates. Analysis of room temperature tensile test data of the single crystal samples reveals that ultimate strength decreases at less than 1 dpa. The copper tungsten laminate tensile tests were completed and fracture analysis is underway. Atom probe analysis of a single crystal tungsten sample irradiated to 2.2 dpa at approximately 750°C revealed rod-like, ribbon-like, and clusters-shaped Os and Re rich precipitates. Tungsten foils irradiated to 2.2 and 3.8 dpa are being prepared for TEM analysis.

4.7 Alloying and Neutron Irradiation Effects in Tungsten-Rhenium — L. M. Garrison (Oak Ridge National Laboratory) **170**

Five alloys of tungsten with different rhenium additions as well as one pure tungsten control material were fabricated by arc melting. After arc melting, the samples were hot-rolled to approximately 80% reduction to improve the microstructure. The distribution of rhenium in one of the alloys was investigated with EDS. Samples for mechanical testing, thermal testing, and neutron irradiation are being prepared from these alloys.

4.8 High-Heat Flux Testing of Low-Level Irradiated Materials Using Plasma Arc Lamps — A.S. Sabau, Y. Katoh, and Sarma Gorti (Oak Ridge National Laboratory) **172**

In this reporting period, a simplified thermo-mechanical model was developed and implemented in ABAQUS™ in order to understand the deformation and residual stresses in the samples tested under high-heat flux. The results of the thermo-mechanical model simulations indicated that W/F82H specimens tested at high-heat fluxes expected in fusion power devices would deform plastically and undergo complex deformation states. The numerical simulation results for the stress levels and pattern of deformation are essential to the understanding of the property degradation and damage mechanisms during HHFT.

5 MAGENTIC AND DIAGNOSTIC SYSTEM MATERIALS

5.1 Irradiation Response of Next Generation High Temperature Superconductor Tapes — K.J. Leonard, F.A. List III, A.M. Williams, J.W. Geringer and T. Aytug (Oak Ridge National Laboratory) **177**

Post-irradiation electrical characterization of two high temperature superconducting tapes (HTS) was performed. The Zr-(Gd,Y)$Ba_2Cu_3O_7$ and (Dy,Y)$Ba_2Cu_3O_7$ tapes are commercially available from SuperPower and American Superconductor, respectively, and are part of a new generation of conductors which utilize nanoparticles and correlated defect structures within the films to produce tapes capable of use under higher applied magnetic fields than earlier $YBa_2Cu_3O_7$ (YBCO) tapes. The results of neutron exposures between 6.54×10^{17} and 1.12×10^{19} n/cm^2 (E>0.1MeV) at irradiation temperatures of 75-80°C, show losses in critical current (I_c) in tests under applied fields up to 0.5 Tesla at 77 K. Increases in I_c loss with dose occurs gradually in the (Dy,Y)$Ba_2Cu_3O_7$ conductor, but is rapid in the Zr-(Gd,Y)$Ba_2Cu_3O_7$ with transport current unable to be measure at fluences above 6.54×10^{17} n/cm^2.

TABLE OF CONTENTS

6 FUSION CORROSION AND COMPATIBILITY SCIENCE

6.1 Liquid Metal Compatibility — S. J. Pawel (Oak Ridge National Laboratory) **185**

Operation of the first thermal convection loop (TCL) using dispersion strengthened FeCrAl (Kanthal APMT) tubing and specimens utilized commercially pure eutectic Pb-17at%Li, and both the peak temperature (550˚C) and temperature gradient (116˚C) were maintained without interruption for 1000 h. The previous report documented limited but somewhat irregular weight change of loop specimens as a function of exposure temperature, with the observation that a pre-oxidation process substantially improved compatibility with the Pb-Li over the exposure period. Further, post-exposure mechanical property data indicated a large increase in strength with concomitant loss of ductility for APMT specimens exposed at or below about 475-500˚C in the TCL. In this reporting period, analysis of TCL specimen cross sections for composition gradients and follow-up tests to examine mechanical properties of APMT exposed for 1000 h in argon (rather than Pb-Li) is documented.

7 MECHANISMS AND ANALYSIS

7.1 Morphology and Mechanical Properties of Commercial $M_{n+1}AX_n$ Phases under Neutron Irradiation — C.Ang, A. Campbell, N. Cetiner, C. Silva, C. Shih, Y. Katoh, S.J. Zinkle (Oak Ridge National Laboratory, USA), T. Toyama, T. Shikama (Tohoku University, Japan) **189**

MAX phases ternary nitrides and carbides were previously characterized (DOE/ER-0313/55). These nominally "Ti_3SiC_2" and "Ti_2AlC" compositions were irradiated to 2 x 10^{25} n/m^2 (E > 0.1 MeV), ~ 2 dpa (SiC). Both materials (referred to as Ti-Si-C and Ti-Al-C) had impurities, including ~6 wt% uncarburized $Al_{11}Ti_5$ and $TiSi_2$ This report covers specimens in Capsules 1, 4 and 7 irradiated at target temperatures of 400, 700 and 1000°C. Mechanical properties of Ti-Al-C were severely degraded at low irradiation temperatures due to grain boundary microcracks attributed to anisotropic swelling. Ti-Si-C consistently maintained moderate strength. Electrical resistivity suggested that metallic Si and Al layers were disrupted at 400°C by point defect accumulation, which recovered in the irradiation temperature interval 400 to 700°C. Ti-Al-C mechanical properties recovered at ~700°C. Young's modulus appeared unaffected. Swelling was more consistent in Ti-Si-C. Conclusions from XRD data were limited, particularly for Ti-Al-C material.

7.2 Ion Irradiation Characterization Studies of MAX Phase Ceramics — D.W. Clark[1], S.J. Zinkle[1,2], M.K. Patel[1], C.M. Parish[2] ([1]University of Tennessee, Knoxville, [2] Oak Ridge National Laboratory) **195**

Ti_3AlC_2, Ti_2AlC and Ti_3SiC_2 MAX phase ceramics were irradiated with 5.8 MeV Ni ions to midrange doses of 10 and 30 dpa at 400 and 700°C. In all cases, the materials remain fully crystalline. X-ray diffraction and nanoindentation show anisotropic swelling and hardening in all materials, with Ti_3AlC_2 and Ti_2AlC exhibiting more pronounced property changes than Ti_3SiC_2. In all three materials there is little damage dependence on dose (suggesting saturation of radiation damage at levels below 10 dpa) and significantly less retained damage at higher temperatures. SEM surface analysis showed significant grain boundary cracking in the aluminum based MAX phase irradiated at 400°C. Ti_3AlC_2 and Ti_2AlC do not

TABLE OF CONTENTS

appear to be suitable for irradiation applications near 400°C whereas the Ti_3SiC_2 is overall more damage tolerant.

7.3 **Effects of Ion and Neutron Irradiation on BAM-11 Bulk Metallic Glass** — J. Brechtl[2], N.A.P Kiran Kumar[1], H. Bei[1], and S. J. Zinkle[1,2] ([1]Oak Ridge National Laboratory, [2]University of Tennessee) **223**

Bulk metallic glasses are candidates for fusion reactor structural components due in part to their good mechanical properties and near net shape fabrication potential. Metallic glasses might also exhibit good radiation resistance as their amorphous structure prohibits the formation of Frenkel defects, and subsequent voids and dislocation loops. Mechanical properties of $Zr_{52.5}Cu_{17.9}Ni_{14.6}Al_{10}Ti_5$ bulk metallic glass (BAM-11) irradiated with fission neutrons to 0.1 and 1 dpa at ~90°C show a slight increase in the Vickers hardness (roughly ~7, ~10% for 0.1, 1 dpa respectively). Using a Nix-Gao method it was found that the nano-indentation hardness increased by about ~0.4% and ~66% for samples irradiated to 0.1 and 1 dpa. The dynamic Young's modulus in the sample irradiated to 0.1 dpa was ~5% less than the control sample, and the nano-indentation Young's modulus at 0.1, 1 dpa was approximately 1.4%, 0.9% less than the control specimen. The density of BAM-11 decreased roughly ~0.4% for both 0.1 and 1 dpa. The results suggest that the irradiation-induced damage begins to saturate at doses between 0.1 and 1 dpa.

7.4 **Exploration of the Radiation Resistance of High Entropy Alloys** — C. Li, S.J. Zinkle (University of Tennessee), N.A.P. Kiran Kumar, H. Bei (Oak Ridge National Laboratory) **230**

This report summarizes the experimental characterization of neutron and ion irradiated 27%Fe-27%Mn-28%Ni-18%Cr HEA. Samples have been neutron irradiated at ~70˚C from 0.1 dpa to 1 dpa, and ion irradiated with 5.8 MeV Ni ions at temperatures ranging from 400˚C to 700˚C and midrange doses from 0.1 dpa to 10 dpa. Post irradiation examination of the neutron irradiated samples found a large increase of hardness after 0.1 dpa, whereas there was only a mild increase from 0.1dpa to 1dpa. Similar trend was observed with the change of yield strength and electrical resistivity. The stress-strain curves of the neutron irradiated specimens are in agreement with the typical behavior of 304, 316 and 347 austenitic stainless steels, but the large yield drop and reduction of work hardening rate seems to be unique to HEA. Ion irradiations at higher temperatures showed evidence of sluggish diffusion with almost no solute depletion or enrichment at grain boundaries. Voids were also not observed at any irradiation condition, suggests that this type of HEA is fairly stable under high temperature radiation.

7.5 **Derivation of True Stress-True Strain Constitutive Laws for Irradiated Ferritic Steels** — T. Yamamoto, G. R. Odette, S. Li (University of California Santa Barbara), S. Maloy, T. Saleh (Los Alamos National Laboratory) **239**

In previous studies we developed a self-consistent approach to derive true stress-stain constitutive [σ(ε)] laws for 8Cr-2W tempered martensitic steel, F82H, encompassing a range of unirradiated and irradiated conditions. In this report, five 8-12Cr tempered martensitic steels and one 14Cr oxide dispersion strengthened alloy, that were side-by-side irradiated to 6.5 dpa at 295°C in the Advanced Test Reactor (ATR), were analyzed. The approach is based on simultaneous measurements and finite element method (FEM) simulations of engineering stress-

TABLE OF CONTENTS

strain s(e) curves, that are consistent with a unique σ(ε) law. In the irradiated condition, the σ(ε) fall into categories of: strain softening, nearly perfectly plastic and strain hardening. Increases in the average σ(ε) in the range of 0-10% strain are smaller than the corresponding increases in the yield stress and vary more from alloy to alloy.

7.6 New Approach to Fracture Mechanics Using Volterra Dislocations, with Applications to Thermo-Mechanical Fracture — A. Sheng, G. Po, N.M. Ghoniem (University of California, Los Angeles) **246**

Complex fracture phenomena involving multiple interacting cracks in three-dimensional geometries have proven difficult to model computationally. This may be a result of the way cracks are typically conceptualized - as a disruption to an otherwise perfect continuum. As a result, existing fracture-modeling methods are generally limited to simple geometries containing a small number of cracks. The mechanical response of the first wall/blanket and divertor systems to plasma transients will inevitably involve analysis of multiple interacting cracks in 3-D geometries. We present here a different strategy for fracture modeling in which cracks are represented by distributions of discrete Volterra dislocations. This new approach overcomes many of the computational difficulties of state-of-the art fracture mechanics based on the Finite Element method.

8 MODELING PROCESSES IN FUSION SYSTEM MATERIALS

8.1 Properties of Vacancy Complexes with H and He in Tungsten from First Principles — G.D. Samolyuk*, Y.N. Osetsky, and R.E. Stoller (Oak Ridge National Laboratory) **255**

It was demonstrated that the formation energy of both single vacancy and six vacancies clusters, the largest cluster investigated in current research, converges for model cells containing 250 atoms. This size is therefore enough to reproduce interaction and formation energy for both H and He atoms. A He atom is strongly attracted to the vacancy with binding energy of 4.6 eV. The di-vacancy clusters are unstable in first nearest (1NN) and second nearest neighbors (2NN) positions and weakly attract each other at 3NN positions. The introduction of H/He stabilizes di-vacancies. The compact 6 vacancies cluster is stable with binding energy of 2 eV.

8.2 Strengthening Due to Radiation Induced Obstacles in Fe and Ferritic Alloys —Yu. N. Osetskiy and R. E. Stoller (Oak Ridge National Laboratory) **259**

Irradiation of structural alloys by neutrons and ions lead to formation of a high density of nanoscale objects such as secondary phase precipitates, voids and gas-filled bubbles. These objects are obstacles for dislocation motion and cause wanted or unwanted changes in mechanical properties. In order to predict materials behavior these obstacles must be characterized as well as their individual strength to be estimated. The only technique that allows this at the scale of nanometers is classical molecular dynamics (MD). In this work we modeled vacancy voids, He-filled bubbles, Cu precipitates and rigid inclusions in bcc-Fe matrix. At the current stage of the research we investigated ½<111>{110} edge dislocation. During its motion this dislocation cannot change its glide plane (cross-slip) to avoid interaction with obstacles and therefore produces maximum strengthening effect. The results obtained in this research will be used to improve theoretical prediction of mechanical properties changes.

TABLE OF CONTENTS

8.3 **Object Kinetic Monte Carlo Simulations of Radiation Damage in Bulk Tungsten** — G. Nandipati, W. Setyawan, H. L. Heinisch, K. J. Roche, R. J. Kurtz (Pacific Northwest National Laboratory) and B. D. Wirth (University of Tennessee) **262**

We used our recently developed lattice based OKMC code; *KSOME* [1] to carry out simulations of radiation damage in bulk W. We study the effect of dimensionality of self-interstitial atom (SIA) diffusion, i.e. 1D versus 3D, on the defect accumulation during irradiation with a primary knock-on atom (PKA) energy of 100 keV at 300 K for dose rates of 10^{-5} and 10^{-6} dpa/s. As expected 3D SIA diffusion significantly reduces damage accumulation due to increased probability of recombination events. In addition, dose rate, over the limited range examined here, appears to have no effect in both cases of SIA diffusion.

8.4 **Quantum Calculations of Energetics of Rhenium Clusters in Tungsten** —W. Setyawan, G. Nandipati, K. J. Roche, R. J. Kurtz (Pacific Northwest National Laboratory), and B. D. Wirth (University of Tennessee) **266**

Density functional theory was employed to explore the energetic properties of clusters up to size 2 of Re in W. While WW<111> is the most stable intrinsic dumbbell, ReW<110> is more stable than ReW<111>. However, when they are trapped by a substitutional Re (Re_s), ReW<111> becomes more stable than ReW<110>. In this case, the most stable configuration forms a ReWRe crowdion with the W atom between the Re atoms. Simulations of a ReW[111] (dumbbell's vector is from Re to W) approaching a Re_s along [111] indicate that the binding energy decreases from 0.83 eV at the first nearest neighbor (NN1) to 0.10 eV at NN3 and ~0 at NN4. In addition, while ReW<111> and ReW<110> are stable near a Re_s at NN1, the ReW<100> instantaneously rotates toward ReW<111>.

8.5 **Displacement Cascade Simulation in Tungsten up to 200 keV of Damage Energy at 300, 1025, and 2050 K** —W. Setyawan, G. Nandipati, K. Roche, R. J. Kurtz (Pacific Northwest National Laboratory) and B. D. Wirth (University of Tennessee, Knoxville) **272**

We generated a molecular dynamics database of primary defects that adequately covers the range of tungsten recoil energy imparted by 14-MeV neutrons. During this semiannual period, cascades at 150 and 200 keV at 300 and 1025 K were simulated. Overall, we included damage energy up to 200 keV at 300 and 1025 K, and up to 100 keV at 2050 K. We report the number of surviving Frenkel pairs (N_F) and the size distribution of defect clusters. The slope of the N_F curve versus cascade damage energy (E_{MD}), on a log-log scale, changes at a transition energy (μ). For E_{MD} > μ, the cascade forms interconnected damage regions that facilitate the formation of large clusters of defects. At 300 K and E_{MD} = 200 keV, the largest size of interstitial cluster and vacancy cluster is 266 and 335, respectively. Similarly, at 1025 K and E_{MD} = 200 keV, the largest size of interstitial cluster and vacancy cluster is 296 and 338, respectively. At 2050 K, large interstitial clusters also routinely form, but practically no large vacancy clusters do.

8.6 **Effects of Temperature on the Flow Stress and Post-Yielding Hardening of Tungsten Micropillars** — Y.N. Cui, G. Po, N.M. Ghoniem (University of California, Los Angeles) **278**

A powerful simulation method is developed to reveal the BCC crystal plastic

TABLE OF CONTENTS

deformation mechanisms. The obtained simulation results are shown to be consistent with available experimental results. In addition, a systematic study is carried out to reveal the new features and mechanisms of plastic flow in tungsten micropillars as a function of temperature.

8.7 **Modeling Microstructural Evolution in Neutron Irradiated Tungsten** — X. Hu, Y. Katoh (Oak Ridge National Laboratory), D. Xu, L.L. Snead, B.D. Wirth (University of Tennessee, Knoxville) **282**

A diffusion-reaction cluster dynamics model based on rate theory was applied to depict the distribution of vacancy and interstitial clusters in tungsten following low temperature (90°C) and low dose (0.03 dpa) neutron irradiation. For experimental data to validate the developed model, *ex situ* characterization of vacancy defects using positron lifetime spectroscopy (PLS) was performed following one-hour anneals at 400, 500, 650, 800, 1000, 1150, and 1300°C for the high purity (110) single crystalline tungsten. TEM observations on tungsten after selected anneal conditions were also performed to determine the dislocation loop population. The data from microstructural observation were then used to validate the developed model as well as the kinetics and energetics parameters describing defect interactions in tungsten obtained from atomistic simulations.

8.8 **Unraveling the Temperature Dependence of the Yield Strength in Single-Crystal Tungsten using Atomistically-Informed Crystal Plasticity Calculations** — David Creceda, Jaime Marian **290**

We use a physically-based crystal plasticity model to predict the yield strength of body-centered cubic (bcc) tungsten single crystals subjected to uniaxial loading. Our model captures the thermally-activated character of screw dislocation motion and full non-Schmid effects, both of which are known to play a critical role in bcc plasticity. The model uses atomistic calculations as the sole source of constitutive information, with no parameter fitting of any kind to experimental data. Our results are in excellent agreement with experimental measurements of the yield stress as a function of temperature for a number of loading orientations. The validated methodology is then employed to calculate the temperature and strain-rate dependence of the yield strength for 231 crystallographic orientations within the standard stereographic triangle. We extract the strain-rate sensitivity of W crystals at different temperatures, and finish with the calculation of yield surfaces under biaxial loading conditions that can be used to define effective yield criteria for engineering design models.

9 **FUSION SYSTEM DESIGN**

9.1 **Preliminary Multi-Physics Design of the First Wall and Blanket System in the Fusion Nuclear Science Facility (FNSF)** —Yue Huang, Nasr Ghoniem (UCLA), Jake Blanchard, Laila El-Guebaly (UW-Madison), Charles Kessel (Princeton Plasma Physics Laboratory), Siegfried Malang (Nuclear Technology Consulting), Mark Tillack (UCSD) **323**

The dual coolant lead-lithium (DCLL) blanket concept is based on a helium-cooled first wall and blanket structure with RAFS (Reduced Activation Ferritic Steel) and a self-cooled Pb-Li breeding zone. 3D solid modeling was achieved with *SOLIDWORKS*, while 3D finite element multiphysics modeling of the DCLL first wall and blanket (mid-plane part of one sector) has been performed by

TABLE OF CONTENTS

COMSOL 5.0 via coupling of the CFD, heat transfer in solids, and heat transfer in fluids modules. Design optimization is realized by seamless connection of two main software platforms with life-links. The results of velocity, pressure, and temperature distributions of helium flow, as well as the primary and secondary thermal stress of the structure were obtained. This is followed by determination of the factors of safety based on the ITER design rules of un-irradiated components. Future efforts will consider the effects of radiation on the lifetime and reliability of the design, improvements in the geometric layout of the FW/B structure, improvements in heat transfer and fluid flow models, and integration of multiscale models of plasticity and fracture.

10 IRRADIATION METHODS, EXPERIMENTS AND SCHEDULES

10.1 Fusion Material Irradiation Test Station (FMITS) at SNS — Mark Wendel, Phil Ferguson (Oak Ridge National Laboratory) **330**

The Fusion Materials Irradiation Test Station (FMITS) is a design concept for installation at the Spallation Neutron Source (SNS) Facility. The project has received funding from OFES during FY15 for (1) performing a mockup test on an FMITS-type target seal, (2) augmenting the safety assessment based on the 2014 technical review, (3) analyzing thermal-hydraulic off-normal transients with a full 3D model to assess the performance of the FMITS sensor array, and (4) remote-handling electrical connector operational mock-up. The main goals of the effort are to remove the project technical risks and prepare for a potential future project.

10.2 HFIR Irradiation Experiments– June 30, 2015 **334**