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1. FERRITIC/MARTENSITIC STEEL DEVELOPMENT

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

The effect of different TMTs on microstructures has been characterized by electron backscattered diffraction (EBSD). The increased interweaving of the microstructures with refined local misorientation zones may be responsible for the improved ductility and Charpy impact resistance. A new RAFM steel has been designed to favor the formation of a high density of MC (M = metal) carbides. The preliminary result of optical images exhibited the presence of a large amount of precipitates in the matrix. Mechanical testing and detailed microstructural characterization of the new alloy will be conducted. Microstructural characterization of the high-dose Fe^{2+} ion irradiated TaC, TaN, and VN specimens has shown different evolutions of the particles as compared to evolution under low-dose irradiation.

1.2 Irradiation Hardening Behavior of F82H and Nickel-Doped F82H - M. Ando, H. Tanigawa, E. Wakai (Japan Atomic Energy Agency), J. P. Robertson and Y. Katoh (Oak Ridge National Laboratory) **4**

For ion irradiation, the irradiation hardening behavior of Nickel-doped F82H steels is similar to that of F82H for 633 K irradiation. However, heavy hardening resulted in Nickel-doped F82H steels irradiated at 543 K. This heavy hardening was especially remarkable after only low dose irradiation. For neutron irradiation, heavy irradiation hardening occurred in the 1.4% Nickel-doped F82H up to 20 dpa; however irradiation hardening of 1.4% Nickel-doped F82H after 80 dpa irradiation is the same level for that of F82H. These results show that the helium effect of mechanical properties in F82H can be examined using F82H and Nickel-doped F82H after high-dose irradiation.

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1.3 On The Effects of Helium-Dpa Interactions on Cavity Evolution in Tempered Martensitic Steels and Nanostructured Ferritic Alloys Under Dual Ion-Beam Irradiation - G. Robert Odette, Takuya Yamamoto, Yuan Wu, (University of California Santa Barbara), Sosuke Kondo, Akihiko Kimura (Kyoto University) **8**

Cavity evolution in normalized and tempered 9Cr martensitic steels (TMS) F82H under Fe^{3+} and He^{+} dual ion beam irradiation (DII) at 500°C was characterized for new dpa and He conditions. The cavity evolution TMS alloy database at 500°C now includes 88 dpa-He-dpa rate DII conditions. The data indicate there may be two swelling trends. This may partly be due to slightly different irradiation flux and/or temperature conditions plus simply natural scatter that is enhanced by local microstructural variations. The fact that swelling under neutron irradiation conditions at low He levels tends to be highly inhomogeneous is well established. The higher swelling rate upper bound data reaches more than ≈ 3% swelling. In the high swelling rate case, the incubation dpa for the onset of void swelling (dpa_i) decrease linearly with increasing He/dpa, and the data can be collapsed on an incubation-adjusted dpa' (= dpa-dpa_i) master trend band. In the low swelling rate case, the incubation dose is ≈ 20 dpa and the post incubation swelling is much more gradual, and approximately linear, with slopes that increase with He/dpa.

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) **14**

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.

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2.2 TEM Characterization of 14YWT and 12YWT ODS Ferritic Alloys Neutron Irradiated at 500°C Using In-Situ Helium Injection - 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) **21**

This report summaries TEM characterization of 14YWT and 12YWT, ODS ferritic alloys with 14 and 12 wt % Cr, respectively, to compare the effect of neutron irradiation with and without concurrent He injection using in situ He injection. The density and average size of <100>/{100} type dislocation loops are always larger than those of 1/2<111>/{111} type, but this difference is significantly affected by He implantation. The density of dislocation loops of both types ranges from ~1 to $4x10^{21}$ m^{-3} with average size ranging from 5~20 nm. 14YWT has lower density but larger size dislocation loops than 12YWT, while the line dislocation density of 14YWT is 3 times lower than that of 12YWT. Helium bubble densities of both 14YWT and 12YWT are $1.9x10^{23}$ m^{-3}, the average He bubbles size of 14YWT and 12YWT are 1.4 and 1.2 nm, respectively. 14YWT exhibits α-α' phase separation, Y-rich particles and uniformly distributed W. In addition to those features, 12YWT exhibits Y-Ti-O particles (not Y-O rich) and elongated Cr-rich phases.

2.3 Irradiation-Damage Behavior in Advanced Steels - C. M. Parish, Y. Katoh (Oak Ridge National Laboratory) and S. Kondo (Kyoto University, Japan) **29**

A 650°C heavy-ion irradiation was performed on 9YWTV, a new 9Cr NFA. This was compared to F82H as a baseline material. Advanced TEM (transmission electron microscopy)-based methods were used to characterize the irradiated zone, and compare to the unirradiated base material. Broadly speaking, little residual radiation damage in the form of black spots or dislocation loops was observed in either the 9YWTV or F82H material, even at areas of the highest dose (>100 dpa). This is likely a result of the high irradiation temperature causing rapid annealing of the irradiation-induced damage. Elemental mapping using EDS (X-ray energy dispersive spectroscopy) or EFTEM (energy filtered TEM) showed evolution of the fine-scale precipitates.

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2.4 Effects of Substrate Etching on Fe - {110} $Y_2Ti_2O_7$ Interfaces; Implications for Nanostructured Ferritic Alloys - T. Stan, Y. Wu, S. Kraemer, and G.R. Odette (University of California Santa Barbara) **35**

Nanostructured Ferritic Alloys (NFAs) have a Fe-Cr matrix and are dispersion strengthened by <5 nm Y-Ti-O phases. The interfaces between these Y-Ti-O nanofeatures (NFs), such as $Y_2Ti_2O_7$ (YTO), and the surrounding matrix provide many favorable properties pertinent to fusion environments, such as trapping He in fine scale bubbles. As a supplement to current characterization efforts of the NFs themselves, surrogate bulk Fe-YTO interfaces have been fabricated by electron beam Fe deposition onto un-etched and etched {110}YTO substrates. The observed Fe-{110}YTO orientation relationships (ORs) and interface characteristics are reported. Etching the substrate prior to Fe deposition has yielded interface structures and ORs that are believed to be similar to those found in embedded NFs. These bulk samples are more amenable to conventional characterization techniques and irradiation experiments. The results also support the development of computational models to predict NFA behavior.

2.5 Tensile Property Characterization of 14YWT Nanostructured Ferritic Alloy NFA1 - M.E. Alam, K. Fields, G. R. Odette (UCSB), D. T. Hoelzer (ORNL) and S. A. Maloy (LANL) **39**

FCRD NFA-1 is a best practice nanostructured ferritic alloy (NFA) produced by ball milling argon atomized Fe-14Cr-3W-0.4Ti-0.2Y (wt.%) and FeO powders, followed by consolidation, extrusion, annealing and cross-rolling. Tensile tests in six orientations at temperatures 23 to 800°C exhibited excellent strength in the extrusion and cross-rolled directions (LT and TL), accompanied by total elongation of ~10%. In contrast, loading in the short plate thickness direction (S) at ambient temperature exhibited much lower strength with a flat, facet fractured surface, and total ductility of less than ~0.5%. The low strength and low total strain in the plate thickness direction can be attributed to the presence of pre-existing microcracks running parallel to the plate surfaces that propagate by brittle cleavage

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2.6 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) **48**

Research during this period focused on (1) designing and developing a FSW process to join 14YWT ODS steel and an ORNL developed 9Cr steel, referred as modified 9Cr steel in this report, (2) dissimilar materials FSW microstructure characterization and flow pattern interpretation, (3) understanding the effect of FSW process conditions on the nano oxide strengthening particles in 14YWT ODS alloy, and (4) CFD based computer modeling to predict temperature fields and reveal flow fields in FSW. We successfully joined 14YWT ODS alloy and modified 9Cr steel without any defects, found FSW temperatures could be high enough to cause the modified 9Cr quenching in the stir zone. The thermal – mechanical process in FSW resulted in nanoparticle coarsening but the process could be finely tuned to preserve nanoclusters with slight coarsening on the 14YWT side of the stir zone. Computer modeling further revealed FSW thermal – mechanical mechanisms.

2.7 Solid-State Mixing of ODS 14YWT and 9Cr F/M Steel Along Butt Joint by Friction Stir Welding - D.T. Hoelzer, L. Tan, W. Tang and Z. Feng (Oak Ridge National Laboratory) **58**

Plates of 5 mm thickness were fabricated from the ODS 14YWT ferritic alloy and 9Cr ferritic/martensitic steel (FMS) and joined by FSW along an effective angle of 9.6º relative to the butt joint between the two plates. Three cross section samples were cut from the joined plates representing solid-state mixtures of ~80%/20%, ~50%/50% and ~20%/80% by volume fraction of 14YWT and 9Cr FMS, respectively and prepared by metallographic procedures. The microstructural analysis of the stir zone observed by Secondary Electron Microscopy (SEM) in the 3 samples showed that very high temperatures from frictional heat caused grain size and oxide particle size increases for 14YWT and decreases in prior austenite grain size and martensite lath size with dissolution of carbides for 9Cr FMS. Complete solid-state mixing occurred in stir zone of the ~50%/50% sample, but only occurred in the region near the pin tool shoulder of the ~80%/20% and ~20%/80% samples.

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3. CERAMIC COMPOSITE STRUCTURAL MATERIAL DEVELOPMENT

3.1 Development of SiC Joining Technologies for Fusion: Test Method Development - T. Koyanagi, J.O. Kiggans, J.M. Pryor, and Y. Katoh (Oak Ridge National Laboratory) **68**

Three different sizes of SiC to SiC brittle joint specimens were prepared for torsional shear tests; the machined hourglass test specimens have necks of 5, 10, or 15 mm diameter at the bonding plane. Uniformity of the bond quality was confirmed by acoustic scanning microscope prior to machining. Pure shear failure was observed for all the specimens, meaning that there was no obvious size effect on the fracture appearance. The size effect on the strength was apparently not clear due to the large scatter in strength values. The causes of the deviations and the possible methods to improve the testing are discussed.

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, R.J. Kurtz, T. Roosendaal, and B. Borlaug; (Pacific Northwest National Laboratory, Richland, WA, USA); M. Ferraris, (Politecnico di Torino, Torino, Italy); A. Ventrella[2], and Y. Katoh, (Oak Ridge National Laboratory) **74**

The use of SiC and SiC-composites in fission or fusion environments requires joining methods for assembling systems. The international fusion community designed miniature torsion specimens for joint testing and irradiation in test reactors with limited irradiation volumes. These torsion specimens fail out-of-plane when joints are strong and when elastic moduli are within a certain range compared to SiC, which causes difficulties in determining shear strengths for joints or for comparing unirradiated and irradiated joints. A finite element damage model was developed that 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. The model was extended to treat elastic-plastic joints such as SiC/epoxy and steel/epoxy joints tested as validation of the specimen design.

3.3 Magnesium Precipitation and Difussion in Mg^+ Ion Implanted Silicon Carbide - W. Jiang, H. J. Jung, L. Kovarik, Z. Wang, T. J. Roosendaal, Z. Zhu, D. J. Edwards, S. Y. Hu, C. H. Henager, Jr., R. J. Kurtz (Pacific Northwest National Laboratory), and Y. Wang (Los Alamos National Laboratory) **85**

Extended abstract of a paper accepted by the Journal of Nuclear Materials.

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3.4 High Dose Irradiation Study of Newer Generation Nuclear Grade SiC Composites: Irradiation Experiments - T. Koyanagi, N.O. Cetiner, Y. Katoh (Oak Ridge National Laboratory), T. Nozawa and K. Ozawa (Japan Atomic Energy Agency) **87**

A study of high dose neutron irradiation effects on newer generation nuclear grade SiC fiber reinforced SiC matrix composites has started, employing a series of new rabbit capsules. The activity is part of the JAEA-USDOE collaboration, with an irradiation matrix designed to achieve a maximum neutron fluence beyond 100 dpa-SiC for a range of irradiation temperature from ~300 to ~1000°C. Twelve capsules containing three types of SiC composites and high-purity monolithic SiC were prepared. Irradiations for all the rabbit capsules will start in Cycle 457 or 458 at the High Flux Isotope Reactor. Properties to be evaluated include dimensional stability, mechanical, thermal, and electrical properties, and microstructures.

4. HIGH HEAT FLUX MATERIALS AND COMPONENT TESTING

4.1 Recent Progress in the Development of Ductile-Phase Toughened Tungsten for Plasma-Facing Materials: W-Ni-Fe Composites - C. H. Henager, Jr., R. J. Kurtz, T. J. Roosendaal, B. A. Borlaug, 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) **93**

A promising approach to increasing fracture toughness and decreasing the DBTT of a W-alloy is by ductile-phase toughening. In this method, a ductile phase is included in a brittle matrix to prevent fracture propagation by crack bridging. W-Ni-Fe alloys consisting of nearly spherical W-particles embedded within a Fe-Ni-W ductile matrix are being manipulated by hot-rolling to create lamellar W/Fe-Ni-W composites with anisotropic fracture properties.

4.2 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), C. H. Henager, Jr., R. J. Kurtz, T. J. Roosendaal, and B. A. Borlaug (Pacific Northwest National Laboratory) **97**

A promising approach to increasing the fracture toughness of W-alloys is ductile-phase toughening (DPT). A ductile phase reinforcement in a brittle matrix increases toughness primarily by crack bridging. A W-Cu laminate was fabricated and the properties of the constituent metals were characterized along with those for the composite. Development of a design model for large-scale crack bridging continued.

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4.3 Process Improvements of Specimen Preparations for Mechanical Testing and Hydrogen Permeation of Nano-W - C. Ren and Z. Zak Fang (University of Utah) **102**

During the last period, attempts have been made to make crack-free specimens for mechanical and hydrogen permeation evaluations of nano W. The ongoing work in this reporting period including: development of a new multi-step pelletizing and compaction process (MPC) for preparing crack-free specimens, and evaluation of the effect of Ti-based additives on the sintering behavior of nano-W materials.

4.4 Fabrication of Functionally Graded, Tungsten Steel Laminate - L.M. Garrison, E. Ohriner (Oak Ridge National Laboratory) **108**

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. A sample of the rolled composite material was polished for metallographic analysis. Metallographic analysis of the foils is ongoing. Samples of the foils and the rolled composite will be tensile tested.

4.5 Recent Progress of Crack Bridging Modeling of Ductile-Phase-Toughened W-Cu Composites – W. Setyawan, C. H. Henager, Jr., K. B. Wagner, T. J. Roosendaal, B. A. Borlaug, And R. J. Kurtz (Pacific Northwest National Laboratory); G. R. Odette, K. H. Cunningham, And F. W. Zok (University Of California, Santa Barbara) **113**

A crack bridging model using calculated Cu stress-strain curves has been developed to study the toughening of W-Cu composites. A strengthening factor and necking parameters have been added to the model for the ductile-phase bridges to incorporate constraint effects at small bridge sizes. Parametric studies are performed to investigate the effect of these parameters. The calculated maximum applied stress intensity, a_{Kmax}, to induce a 1-mm stable crack is compared to the experimental stress intensity at peak load, K_{peak}. Without bridge necking, increasing the strengthening factor improves the agreement between a_{Kmax} and K_{peak} when plotted vs. logarithm of the displacement rate. Improvement can also be achieved by allowing necking with a larger failure strain. While the slope is better matched with this latter approach, the calculated value of a_{Kmax} is significantly larger than K_{peak}.

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4.6 Alloying and Neutron Irradiation Effects in Tungsten-Rhenium - L.M. Garrison, L.L. Snead, E. Ohriner (Oak Ridge National Laboratory) **120**

Five alloys of tungsten with different amounts of 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. Samples for mechanical testing, thermal testing, and neutron irradiation are being prepared from these alloys.

4.7 Neutron Irradiation Effects in Tungsten - L.M. Garrison, L.L. Snead, X. Hu, T. Koyanagi, N.A.P. Kiran Kumar (Oak Ridge National Laboratory), M. Fukuda (Tohoku University, Japan) and K. Yabuuchi (Tohoku University, Japan) **122**

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-copper laminates. Each rabbit irradiation capsule also contained silicon carbide (SiC) samples to be used as temperature monitors. The post-irradiation analysis of the SiC samples has been completed, and the estimated tungsten irradiation temperatures were determined. A subset of single crystal [110] samples was selected to study the effects of annealing on recovering mechanical properties and defects through positron annihilation spectroscopy (PAS) and hardness testing.

4.8 High-Heat Flux Testing of Low-Level Irradiated Materials Using Plasma Arc Lamps - A.S. Sabau, L.L. Snead, Y. Katoh (Oak Ridge National Laboratory), Kazutoshi Tokunaga (Kyushu University, Japan) **127**

In this reporting period, high-heat flux testing of five specimens, supplied by the Japanese collaborators in the PHENIX program, was conducted at ORNL. Dr. Tokunaga was on assignment at ORNL for four weeks as part of the joint US/Japan PHENIX program, during which time this high-heat flux testing and evaluation of the clamping fixture and vacuum system was conducted.

5. MAGNETIC AND DIAGNOSTIC SYSTEM MATERIALS –

No contributions this period.

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6. FUSION CORROSION AND COMPATIBILITY SCIENCE

6.1 Liquid Metal Compatibility - S. J. Pawel (Oak Ridge National Laboratory, USA) **132**

Operation of the first thermal convection loop (TCL) using dispersion strengthened FeCrAl (Kanthal APMT) tubing and specimens was completed in the previous reporting period. The working fluid in the TCL was eutectic Pb-17at%Li, and both the peak temperature (550˚C) and temperature gradient (116˚C) were maintained without interruption for 1000 h. When TCL operation was terminated, a problem with complete draining of the Pb-Li from the loop was encountered, which delayed retrieval of specimens and related examinations; however, post-exposure analysis of all specimens is now nearing completion.

7. MECHANISMS AND ANALYSIS

7.1 The Origin of Strain Avalanches in Submicron Plasticity in fcc Metals - T. Crosby, G. Po, C. Erel, N.M. Ghoniem (University of California, Los Angeles) **136**

Extended abstract of a paper accepted with revisions in Acta Materialia (2014).

7.2 Nudged Elastic Band Simulations of Kink Pairs in Tungsten - David Cereceda and Jaime Marian (Lawrence Livermore National Laboratory and University of California at Los Angeles) **138**

Atomistic techniques have been used to calculate energy barriers for dislocation motion that control the strength (yield stress and flow stress) of the material. In particular, the calculations focus on the change in enthalpy as a straight dislocation moves through the crystal lattice (the Peierls barrier) and kink pair formation enthalpy that controls the thermally activated double-kink mechanism important at low to moderate stresses. A novel means of assessing kink widths within atomistic simulations is introduced.

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8. MODELING PROCESSES IN FUSION SYSTEM MATERIALS

8.1 Molecular Dynamics of Cascades Overlap in Tungsten with 20-keV Primary Knock-On Atoms - W. Setyawan, G. Nandipati, K. J. Roche, R.J. Kurtz (Pacific Northwest National Laboratory[1]); B. D. Wirth (University of Tennessee) **156**

Molecular dynamics simulations are performed to investigate the mutual influence of two subsequent cascades in tungsten. The influence is studied using 20-keV primary knock-on atoms, to induce one cascade after another separated by 15 ps, in a lattice temperature of 1025 K (i.e. 0.25 of the melting temperature of the interatomic potential). The center of mass of the vacancies at the peak damage during the cascade is taken as the location of the cascade. The distance between this location to that of the next cascade is taken as the overlap parameter. Empirical fits describing the number of surviving vacancies and interstitial atoms as a function of overlap are presented.

8.2 Object Kinetic Monte Carlo Simulations of Radiation Damage in 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) **160**

We used our recently developed lattice-based object kinetic Monte Carlo code; *KSOME* [1] to carryout simulations of radiation damage in bulk tungsten at temperatures of 300, and 2050 K for various dose rates. Displacement cascades generated from molecular dynamics (MD) simulations for PKA energies at 60, 75 and 100 keV provided residual point defect distributions. It was found that the number density of vacancies in the simulation box does not change with dose rate while the number density of vacancy clusters slightly decreases with dose rate indicating that bigger clusters are formed at larger dose rates. At 300 K, although the average vacancy cluster size increases slightly, the vast majority of vacancies exist as mono-vacancies. At 2050 K no accumulation of defects was observed during irradiation over a wide range of dose rates for all PKA energies studied in this work.

8.3 Singularity-Free Dislocation Dynamics with Strain Gradient Elasticity - G. Po, M. Lazar, D. Seif, N.M. Ghoniem (University of California, Los Angeles) **166**

Extended abstract of a paper published in JMPS 68 (2014) 161-178.

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8.4 **Atomic-Scale Mechanisms of Helium Bubble Hardening in Iron -** Yuri N. Osetsky and Roger E. Stoller (Oak Ridge National Laboratory) **168**

Generation of helium in (n, α) transmutations changes the response of structural materials to neutron irradiation. The whole process of radiation damage evolution is affected by He accumulation and leads to significant changes in the material properties. A population of nanometer-size He-filled bubbles affects mechanical properties and the impact can be quite significant at high bubble density. Understanding how these basic mechanisms affect mechanical properties is necessary for predicting radiation effects. We completed an extensive study of the interactions between a moving edge dislocation and bubbles using atomic-scale modeling, focused on the effect of bubble size and He concentration in bubbles. The ability of bubbles to act as obstacles to dislocation motion is close to that of voids when the He-to-vacancy ratio is in the range 0 to 1. A few simulations at higher He contents demonstrated that the interaction mechanism is changed for over-pressurized bubbles and they become weaker obstacles. The results are discussed in light of post-irradiation materials testing.

8.5 **Molecular Dynamics Modeling of Atomic Displacement Cascades in 3C-SiC: Comparison of Interatomic Potentials -** G. D. Samolyuk, Y. N. Osetskiy and R. E. Stoller (Oak Ridge National Laboratory) **178**

Molecular dynamics modeling of cascades characterized the primary radiation damage in 3C-SiC, demonstrating that commonly used interatomic potentials are inconsistent with *ab initio* calculations of defect energetics. Both the Tersoff potential and a modified embedded-atom method potential reveal a barrier to recombination of the carbon interstitial and carbon vacancy much higher than the density functional theory (DFT) results. The barrier obtained with a newer Gao and Weber potential is closer to the DFT result, with significant differences in point defect production. Both 10 and 50 keV cascades were simulated in 3C-SiC at a range of temperatures. In contrast to the Tersoff potential, the Gao-Weber potential produces almost twice as many C vacancies and interstitials at the time of maximum disorder (~0.2 ps) but only about 25% more stable defects at the end of the simulation. Only about 20% of the carbon defects produced with the Tersoff potential recombine during the in-cascade annealing phase, while about 60% recombine with the Gao-Weber potential. Gao-Weber appears to give a more realistic description, with some shortcomings compared to the *ab initio* results.

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9. IRRADIATION METHODS, EXPERIMENTS AND SCHEDULES

9.1 Fusion Material Irradiation Test Station (FMITS) at SNS - Mark Wendel, Phil Ferguson (Oak Ridge National Laboratory) **188**

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, 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 upcoming viability review by DOE.

9.2 HFIR Irradiation Experiments – December 31, 2014 **192**