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- 3.2 **ANALYSIS OF Ta-RICH MX PRECIPITATES IN RAFS**—H. Tanigawa (Japan Atomic Energy Research Institute), H. Sakasegawa (Kyoto University), N. Hashimoto, S. J. Zinkle, and R. L. Klueh (Oak Ridge National Laboratory), and A. Kohyama (Kyoto University) 33
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Transmission electron microscopy (TEM) specimens of F82H-IEA, F82H HT2, JLF-1, and ORNL9Cr steels were prepared from miniature Charpy specimens irradiated up to 5 dpa at 573K by using the Focused Ion Beam (FIB) technique in order to determine the mechanisms that cause the difference in Charpy impact properties. TEM microstructural analysis was performed with emphasis on dislocation structure and precipitate distribution. The TEM specimens indicated no significant difference on dislocation microstructures, such as dislocation loop size and density, in the steels. While precipitate's distribution of each steel was somewhat different in their size and density, larger precipitates were observed on prior austenite grain (PAG) boundaries and martensite packet boundaries of F82H-IEA and F82H HT2 compared to JLF-1 and ORNL9Cr. TEM analysis also suggested that ORNL9Cr had the finest grain structure, and F82H had a coarse grain structure. The microstructure of the deformed region of irradiated F82H-IEA contained dislocation channels. This suggests that dislocation channeling could be the dominant deformation mechanism in the RAFs, resulting in the loss of strain-hardening capacity.

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 D. S. Gelles (Pacific Northwest National Laboratory) and Mikhail A. Sokolov (Oak Ridge National Laboratory)

The fracture toughness database for F82H displays some anomalous behavior associated with the center of the 25 mm thick plate. Metallographic carbide etchant reveals larger particles dispersed through the 25 mm thick F82H plate. The particles are found to be rich in Ta and O. Size distribution measurements indicate no enhancement at the center of the plate. However, the spatial distribution is affected so that large particles are more often located next to other large particles in the center of the plate. A mechanism is proposed that promotes easy crack nucleation between large tantalum oxide particles.

- 3.6 A MASTER CURVE ANALYSIS OF F82H USING STATISTICAL AND CONSTRAINT LOSS SIZE ADJUSTMENTS OF SMALL SPECIMEN DATA—** 51
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We assembled a fracture toughness database for the IEA heat of F82H based on a variety of specimen sizes with an ASTM E1921 nominal master curve (MC) reference temperature, $T_0 = -119 \pm 3^\circ\text{C}$. However, the data are not well represented by a MC. T_0 decreases systematically with a decreasing deformation limit M_{lim} starting at $\sigma_0/200$, which

is much higher than the E1921 censoring limit of 30, indicating large constraint loss in small specimens. The small scale yielding T_0 at high Mlim is $-98 \pm 5^\circ\text{C}$. While, the scatter is somewhat larger than predicted, after model-based adjustments for the effects of constraint loss, the data are in reasonably good agreement with a MC with a $T_0 = -98^\circ\text{C}$. This supports to use of MC methods to characterize irradiation embrittlement, as long as both constraint loss effects are properly accounted for. Finally, we note various issues, including sources of the possible excess scatter, which remain to be fully assessed.

- 3.7 ON THE EFFECTS OF FATIGUE PRECRACKING ON THE MICROSTRUCTURE AROUND PRECRACK IN 1TCT FRACTURE TOUGHNESS SPECIMEN OF F82H-IEA—H. Tanigawa (Japan Atomic Energy Research Institute), N. Hashimoto, M. A. Sokolov, and R. L. Klueh (Oak Ridge National Laboratory), and M. Ando (JAERI) 58**

1TCT fracture toughness specimens of F82H-IEA steel were fatigue precracked and sliced in specimen thickness wise for microstructure analysis around the precrack. The microstructure around the precrack was observed by optical microscopy (OM), scanning electron microscopy (SEM), orientation imaging microscopy (OIM), and transmission electron microscopy (TEM). TEM samples around the crack front were prepared by focused ion beam (FIB) processor. The fracture surfaces of tested 1TCT specimens were also observed. OM observation showed that the precrack penetration was straight in the beginning, and then tended to follow a prior austenite grain boundary and to branch into 2 to 3 directions at the terminal. SEM and OIM observations revealed that the both microstructures around the precracks and ahead of the precrack had turned into cell structure, which is the typical microstructure of fatigue-loaded F82H. TEM images and inverse pole figures obtained from the crack-front region confirmed this structure change. Possible mechanisms by which the precrack branching or the cell structure ahead of precracks affects fracture toughness were suggested.

- 3.8 TEM OBSERVATION AROUND CRACK IN FATIGUE-PRECRACKED 1TCT FRACTURE TOUGHNESS SPECIMEN OF F82H-IEA—N. Hashimoto (Oak Ridge National Laboratory), H. Tanigawa (Japan Atomic Energy Research Institute), M. Ando (JAERI), and M. A. Sokolov (ORNL) 67**

Transmission electron microscopy (TEM) specimens of F82H-IEA were prepared from a middle section of the fatigue-precracked 1TCT specimens and fabricated by using the Focused Ion Beam (FIB) technique in order to investigate microstructural evolution with crack propagation. The TEM specimens, taken from the area around crack, the area of crack tip, and the area in ahead of the crack tip, indicated the presence of cell structure that was generally seen in fatigue-loaded ferritic/martensitic steels. It is possible that this cell structure affects the fracture toughness, however, the effect would be negligible for irradiated specimen due to elimination of the cell structure during irradiation.

- 3.9 IRRADIATION CREEP AND SWELLING FROM 400°C TO 600°C OF THE OXIDE DISPERSION STRENGTHENED FERRITIC ALLOY MA957—M. B. Toloczko, D. S. Gelles, F. A. Garner, and R. J. Kurtz (Pacific Northwest National Laboratory),* and K. Abe (Dept. of Quantum Sci. and Energy Eng., Tohoku University, Sendai, Japan) 71**

Extended Abstract

- 3.10 IRRADIATION EFFECTS ON TENSILE PROPERTIES OF HIGH-CHROMIUM FERRITIC/MARTENSITIC STEELS—R. L. Klueh (Oak Ridge National Laboratory) 73**

Tensile specimens of four ferritic/martensitic steels were irradiated at 390-395°C in the Experimental Breeder Reactor (EBR-II) to 32-33 dpa. The steels were the ORNL reduced-activation 9Cr-2WV/Ta and that steel containing 2% Ni (9Cr-2WV/Ta-2Ni), modified 9Cr-1Mo, and Sandvik HT9 (12Cr-1MoVW). The 9Cr-2WV/Ta and 9Cr-2WV/Ta-Ni were irradiated after normalizing and tempering some specimens 1 hr at 700°C and

some specimens 1 h at 750°C; the 9Cr-1MoVNb and 12Cr-1MoVW were tempered 1 h at 760°C. Based on the change in tensile properties, the results demonstrated the superiority of the 9Cr-2WVTa steel over the two commercial steels. Charpy properties of the 9Cr-2WVTa-2Ni steel were similar to those of the 9Cr-2WVTa steel, indicating no adverse effect of the nickel on the properties after irradiation in a fast reactor around 400°C.

- 3.11 AN ANALYSIS OF THE EFFECTS OF HELIUM ON FAST FRACTURE AND EMBRITTLEMENT OF \approx 8Cr TEMPERED MARTENSITIC STEELS**—G. R. Odette, T. Yamamoto, and H. Kishimoto (University of California, Santa Barbara) **80**

We have assembled the available paired datasets on irradiation-induced increases in yield stress ($\Delta\sigma_y$) and transition temperature shifts (ΔT), in order to assess the potential role of high levels of helium on irradiation embrittlement of \approx 8Cr martensitic steels. Both ΔT versus $\Delta\sigma_y$ scatter plots and variations in the hardening-shift coefficient, $C = \Delta T / \Delta\sigma_y$, are used to evaluate potential non-hardening helium embrittlement (NHHE) contributions to ΔT . The available data is limited, scattered, and potentially confounded. However, collectively the database suggests that there is a minimal NHHE up to a few hundred appm. However, a NHHE contribution appears to emerge at higher helium concentrations, estimated to be more than 400 to 600 appm. The NHHE is accompanied by a transition from transgranular cleavage (TGC) to intergranular fracture (IGF). IGF generally occurs only at high $\Delta\sigma_y$. Synergistic combinations of large $\Delta\sigma_y$ and severe NHHE could lead to very large ΔT in first wall and blanket structures at fusion spectrum dose levels above 50 to 75 dpa. Future research will focus on continued collection and analysis of data, participation in new experiments to better address NHHE and developing detailed micromechanical models of helium effects.

- 3.12 INFLUENCE OF HIGH DOSE NEUTRON IRRADIATION ON MICROSTRUCTURE OF EP-450 FERRITIC-MARTENSITIC STEEL IRRADIATED IN THREE RUSSIAN FAST REACTORS**—A. M. Dvoriashin, S. I. Porollo, and Yu. V. Konobeev (Institute of Physics and Power Engineering), and F. A. Garner (Pacific Northwest National Laboratory) **91**

The microstructure of EP-450 ferritic-martensitic steel was determined after irradiation in BN-350, BOR-60 and BR-10 fast reactors at temperatures in the range 275-690°C. The examinations confirm a high resistance of EP-450 steel to void swelling, but the resistance appears to be lower when the dpa rate is reduced. Depending on irradiation dose and temperature the following was observed: voids (285-520°C), dislocation loops and linear dislocations (275-520°C), α' -phase (285-520°C), χ phase (460-590°C), and M_2X precipitates (460-690°C). It appears that the formation of dislocation loops and α' precipitates at high densities is responsible for the low temperature embrittlement observed in this steel.

- 3.13 COMPILATION AND PRELIMINARY ANALYSIS OF A IRRADIATION HARDENING AND EMBRITTLEMENT DATABASE FOR 8Cr MARTENSITIC STEELS**—T. Yamamoto, G. R. Odette, H. Kishimoto (University of California, Santa Barbara), and J. W. Rensman (NRG, Petten) **100**

Data on irradiation hardening and embrittlement of 7-9Cr normalized and tempered martensitic steels (TMS) has been compiled from the literature, including results from neutron, spallation proton (SP) and He-ion (HI) irradiations. Limitations of this database are briefly described. Simple, phenomenological-empirical fitting models were used to assess the dose (displacement-per-atom, dpa), irradiation temperature (T_i) and test temperature (T_t) dependence of yield stress changes ($\Delta\sigma_y$), as well as the corresponding dependence of sub-sized Charpy V-notch impact test transition temperature shifts (ΔT_c). The $\Delta\sigma_y$ are similar for SP and neutron irradiations, with very high and low helium to dpa ratios, respectively. The $\Delta\sigma_y$ trends were found to be remarkably consistent with the T_i and dpa hardening-dependence of low alloy steels irradiated at much lower doses. The

some specimens 1 h at 750°C; the 9Cr-1MoVNb and 12Cr-1MoVW were tempered 1 h at 760°C. Based on the change in tensile properties, the results demonstrated the superiority of the 9Cr-2WVTa steel over the two commercial steels. Charpy properties of the 9Cr-2WVTa-2Ni steel were similar to those of the 9Cr-2WVTa steel, indicating no adverse effect of the nickel on the properties after irradiation in a fast reactor around 400°C.

- 3.11 AN ANALYSIS OF THE EFFECTS OF HELIUM ON FAST FRACTURE AND EMBRITTLEMENT OF \approx 8Cr TEMPERED MARTENSITIC STEELS**—G. R. Odette, T. Yamamoto, and H. Kishimoto (University of California, Santa Barbara) **80**

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- 3.12 INFLUENCE OF HIGH DOSE NEUTRON IRRADIATION ON MICROSTRUCTURE OF EP-450 FERRITIC-MARTENSITIC STEEL IRRADIATED IN THREE RUSSIAN FAST REACTORS**—A. M. Dvoriashin, S. I. Porollo, and Yu. V. Konobeev (Institute of Physics and Power Engineering), and F. A. Garner (Pacific Northwest National Laboratory) **91**

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- 3.13 COMPILATION AND PRELIMINARY ANALYSIS OF A IRRADIATION HARDENING AND EMBRITTLEMENT DATABASE FOR 8Cr MARTENSITIC STEELS**—T. Yamamoto, G. R. Odette, H. Kishimoto (University of California, Santa Barbara), and J. W. Rensman (NRG, Petten) **100**

Data on irradiation hardening and embrittlement of 7-9Cr normalized and tempered martensitic steels (TMS) has been compiled from the literature, including results from neutron, spallation proton (SP) and He-ion (HI) irradiations. Limitations of this database are briefly described. Simple, phenomenological-empirical fitting models were used to assess the dose (displacement-per-atom, dpa), irradiation temperature (T_i) and test temperature (T_t) dependence of yield stress changes ($\Delta\sigma_y$), as well as the corresponding dependence of sub-sized Charpy V-notch impact test transition temperature shifts (ΔT_c). The $\Delta\sigma_y$ are similar for SP and neutron irradiations, with very high and low helium to dpa ratios, respectively. The $\Delta\sigma_y$ trends were found to be remarkably consistent with the T_i and dpa hardening-dependence of low alloy steels irradiated at much lower doses. The

similar T_i and (low) dose dependence of $\Delta\sigma_y$ and ΔT_c , as well as an analysis of paired ΔT_c - $\Delta\sigma_y$ datasets, show that embrittlement is dominated by a hardening mechanism below about 400°C. However, the corresponding hardening-Charpy shift coefficient, $C_c = \Delta T_c / \Delta\sigma_y \approx 0.38 \pm 0.18$ is lower than that for the fracture toughness reference temperature ($\approx 0.57 \pm 0.1$), indicating that sub-sized Charpy tests provide *non-conservative* estimates of embrittlement. The C_c increases, or sometimes even takes on negative values, at $T_i \geq 400^\circ\text{C}$, indicative of a non-hardening embrittlement (NHE) contribution. Analysis of limited data on embrittlement due to thermal aging supports this conclusion, and we hypothesize that the NHE regime is shifted to lower temperatures by radiation enhanced diffusion. Possible effects of helium on embrittlement are addressed in a companion report.

- 3.14 EXTRAPOLATION OF FRACTURE TOUGHNESS DATA FOR HT9 IRRADIATED AT TEMPERATURES 360-390°C**—R. J. Kurtz and D. S. Gelles (Pacific Northwest National Laboratory) **115**

Following irradiation in the AC01 test at 360°C to 5.5×10^{22} n/cm², two HT9 samples tested at 30°C were found to have fracture toughness levels of 28.2 and 31.9 MPa m^{1/2}, whereas a third identical specimen tested at 205°C gave 126 MPa m^{1/2}. Based on testing of notched tensile specimens from the same irradiation test, the low toughness was a result of brittle fracture. A similar low level of toughness has also been demonstrated in HT9 following irradiation at 250°C and therefore such behavior is reproducible.

- 3.15 NANOFEATURE DEVELOPMENT AND STABILITY IN NANOSTRUCTURED FERRITIC ALLOYS**—M. J. Alinger and G. R. Odette (University of California, Santa Barbara) and D. T. Hoelzer (Oak Ridge National Laboratory) **129**

Ferritic alloys containing a high density of nanoscale clusters of Y-Ti-O exhibit superior creep strength as well as potential for high resistance to radiation damage. Small angle neutron scattering (SANS) was used to characterize the sequence-of-events and the necessary ingredients for the formation of nanoclusters (NCs) during processing, as well as their thermal stability during high temperature aging. Mechanical alloying (MA) dissolves Y₂O₃ in the master alloy Fe-Cr-W powders. A large population of 1-2 nm NCs precipitate during subsequent high temperature consolidation. The NC sizes increase and their volume fractions and number densities decrease with increasing the consolidation temperature. Both Ti and Y are necessary for NC formation at higher temperatures. The NCs in MA957 are stable during aging at 1150°C for times up to 243 h, but systematically coarsen at 1200°C. The NCs coarsen rapidly and become unstable at higher aging temperatures. Variations in the alloy hardness are consistent with differences in the NC sizes and number densities.

- 4.0 COPPER ALLOYS** **135**

- 4.1 EVOLUTION OF CLEARED CHANNELS IN NEUTRON-IRRADIATED PURE COPPER AS A FUNCTION OF TENSILE STRAIN**—D. J. Edwards (Pacific Northwest National Laboratory) and B. N. Singh (Risø National Laboratory, Denmark) **136**

Extended Abstract

- 4.2 DISLOCATION DENSITY-BASED CONSTITUTIVE MODEL FOR THE MECHANICAL BEHAVIOR OF IRRADIATED CU**—A. Arsenlis (Lawrence Livermore National Laboratory), B. D. Wirth (Department of Nuclear Engineering, University of California Berkeley), and M. Rhee (Lawrence Livermore National Laboratory) **139**

Extended Abstract

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	Recent irradiation experiments conducted on a variety of austenitic stainless steels have shown that void swelling appears to be increased when the dpa rate is decreased, primarily by a shortening of the transient regime of swelling. This paper presents results derived from nominally similar irradiations conducted on six Russian steels, all laboratory heat variants of Fe-16Cr-15Ni-3Mo-Nb-B, with each irradiated in two fast reactors, BOR-60 and BN-350. The BN-350 irradiation proceeded at a dpa rate three times higher than that conducted in BOR-60. In all six steels, a significantly higher swelling level was attained in BOR-60, agreeing with the results of earlier studies.	
6.2	INFLUENCE OF RADIATION-INDUCED VOIDS AND BUBBLES ON PHYSICAL PROPERTIES OF AUSTENITIC STRUCTURAL ALLOYS—E. N. Shcherbakov, A. V. Kozlov, and I. A. Portnykh (FSUE Institute of Nuclear Materials), Iouri I. Balachov (SRI International), and F. A. Garner (Pacific Northwest National Laboratory)	147
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Inorganic Materials, Russia)

There are very few candidate MHD coating materials since Li dissolves most oxides and many carbides and nitrides do not have sufficient electrical resistivity for this application. The past few years have seen great changes in the research emphasis and strategy for MHD coatings. Problems with CaO have led to a focus on new candidates with low cation solubility in Li, such as Y_2O_3 and Er_2O_3 . Coatings of these materials are being fabricated by a variety of processing techniques and the resistivity and microstructure characterized. Progress is being made in the development of MHD coatings, but as yet no coatings have shown sufficient compatibility with Li. Electrical resistivity results from Y_2O_3 coatings as-deposited and after exposure to Li are presented. Self-healing and in-situ coatings are being investigated based on CaO from Li-Ca and Er_2O_3 from Li-Er. Anticipated problems with defects in ceramic coatings, either as-fabricated or due to tensile cracking, suggests that the most viable coating strategy will have to be multi-layered. An outer metallic layer will prevent Li from wetting cracks in the inner ceramic insulating layer and also limit interaction between the ceramic and Li. Whether the MHD coating is single- or dual-layered, processing issues will need to be addressed before the issue of compatibility can be answered.

8.0 BREEDING MATERIALS 170

No contributions

9.0 RADIATION EFFECTS, MECHANISTIC STUDIES, AND EXPERIMENTAL METHOD 171

9.1 THE EFFECTS OF INTERFACES ON RADIATION DAMAGE PRODUCTION IN LAYERED METAL COMPOSITES—H. L. Heinisch, F. Gao, and R. J. Kurtz (Pacific Northwest National Laboratory) 172

Multilayered composites consisting of many alternating metal layers, each only nanometers thick, possess enormous strength, approaching theoretical limits. These materials also display unexpectedly high thermal and mechanical stability [1]. Their unique properties derive from the operation of deformation mechanisms that do not occur in conventional metallic materials and are a result of the large internal interfacial areas and high coherency strains of the nanolayered metals. The enormous interface area to volume ratio of these materials may also positively affect their resistance to radiation damage, making them potentially useful materials for applications in fusion reactors.

9.2 MULTISCALE MODELING OF RADIATION DAMAGE IN Fe-BASED ALLOYS IN THE FUSION ENVIRONMENT—B. D. Wirth (Department of Nuclear Engineering, University of California Berkeley), G. R. Odette (University of California, Santa Barbara), J. Marian (California Institute of Technology), L. Ventelon (University of California, Berkeley), J. A. Young and L. A. Zepeda-Ruiz (Lawrence Livermore National Laboratory) 179

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9.3 INFLUENCE OF PKA DIRECTION, FREE SURFACES, AND PRE-EXISTING DEFECTS ON CASCADE DAMAGE FORMATION—R. E. Stoller, S. G. Guiriec (Oak Ridge National Laboratory) 180

Primary cascade damage production in iron has been extensively investigated by molecular dynamics, and average defect production parameters, such as the total number of stable point defects, in-cascade defect clustering fractions, and in-cascade cluster size distributions have been derived. However, preliminary results indicated several factors could alter "normal" cascade evolution and lead to quite different defect production behavior. Further investigations of three such factors have been carried out: (1) primary knock-on atom (PKA) direction, (2) nearby free surfaces, and (3) pre-existing effects.

Results of the investigation confirm these factors can significantly impact cascade damage formation. The effects include enhanced defect survival for PKA directions that lie in close-packed {110} planes, increased point defect clustering and larger defect clusters for cascades initiated near a surface, and reduced defect survival in simulation cells containing defects. The origin and implications of these effects are discussed relative to the interpretation of certain experimental observations and parameters used in other modeling studies.

- 9.4 DISLOCATION INTERACTIONS WITH VOIDS AND HELIUM BUBBLES IN FCC METALS**—J. A. Young (Lawrence Livermore National Laboratory), B. D. Wirth (Department of Nuclear Engineering, University of California Berkeley), J. Robach, and I. M. Robertson (University of Illinois, Urbana-Champaign) **190**

Extended Abstract

- 9.5 THE EFFECTS OF GRAIN BOUNDARY STRUCTURE ON BINDING OF He IN Fe**—R. J. Kurtz and H. L. Heinisch, Jr. (Pacific Northwest National Laboratory) **191**

Extended Abstract

- 9.6 DISLOCATION-STACKING FAULT TETRAHEDRON INTERACTION: WHAT WE CAN LEARN FROM ATOMIC SCALE MODELING**—Yu. N. Osetsky, R. E. Stoller, and Y. Matsukawa (Oak Ridge National Laboratory) **195**

Stacking fault tetrahedra (SFTs) are formed under irradiation of fcc metals and alloys with low stacking fault energy. The high number density of SFTs observed suggests that they should contribute to radiation-induced hardening, and, therefore, taken into account when estimating mechanical properties changes of irradiated materials. The central issue is describing the individual interaction between a moving dislocation and an SFT, which is characterized by a very fine size scale on the order of a few to one-hundred nanometers. This scale is amenable to both in-situ TEM experiments and large-scale atomic modeling. In this paper we present results of an atomistic simulation of dislocation-SFT interactions using molecular dynamics (MD). The MD simulations modeled an edge dislocation interacting with SFTs with different sizes and at different temperatures and strain rates. The results are compared with observations from in-situ deformation experiments in which several interactions between moving dislocations and SFTs were observed. It is demonstrated that in some cases the simulations and experimental observations are quite similar, suggesting a reasonable interpretation of experimental observations. Other cases, when modelling does not reproduce experimental observations, are also discussed and the importance of strain rate, dislocation nature and specimen surface effect are indicated.

- 9.7 KINETIC MONTE CARLO SIMULATIONS OF DISLOCATION DECORATION AND RAFT FORMATION IN BCC-IRON UNDER CASCADE IRRADIATION**—M. Wen, N. M. Ghoniem (Department of Mechanical and Aerospace Engineering, University of California, Los Angeles), and B. N. Singh (Risø National Laboratory, Denmark) **201**

Under neutron irradiation, primary defect clusters of both self-interstitial atom (SIA) and vacancy type are directly produced in displacement cascades, which have been confirmed by experiments as well as molecular dynamics simulations. The highly mobile SIA clusters play a crucial role in the development of characteristic microstructures, such as rafts of SIA clusters and dislocation decoration, and the corresponding radiation hardening behavior [1]. We have developed a new approach to KMC simulations to investigate the segregation and accumulation of point defects at the atomic scale with incorporating the elastic interaction between defect clusters and microstructures by using the elastic representation of point defects due to Kröner [2]. The decoration of dislocations by SIA clusters and the formation of rafts in bcc-iron are modeled in detail, and the general

conditions for the occurrence and development of both features have are discussed. We also present results of SIA cluster density as a function of irradiation dose, using cluster size distributions, cascade frequency, and individual cluster dynamic parameters obtained from molecular dynamics simulations. Good agreement is found between the results of our present KMC simulations and experimental observations. The one-dimensional motion of glissile SIA clusters and the interaction between defects and microstructures are shown to be the main cause for appearance and development of characteristic microstructures. It is demonstrated that KMC computer simulation is a valuable tool in studying defect kinetics and microstructure evolution, taking into account many different microscopic mechanisms and covering very different time and length scales.

- 9.8 MODELING THE BRITTLE-DUCTILE TRANSITION IN FERRITIC STEELS—S. J. Noronha and N. M. Ghoniem (University of California, Los Angeles) 209**

The crack tip plastic zone is represented using an array of dislocations emitted from the crack-tip plasticity on loading. The dislocations emitted shield the crack-tip, thereby enhancing the applied stress intensity for fracture from the Griffith value. The stress intensity at fracture is thus a function of the dislocation distribution around the crack tip. This distribution in effect is controlled by the mobility and nucleation energy of dislocations. The method is used to simulate the case where microcracks in the plastic zone of the macrocrack initiate cleavage fracture.

- 9.9 MD AND KMC MODELING OF THE GROWTH AND SHRINKAGE MECHANISMS OF HELIUM-VACANCY COMPLEXES IN FE—K. Morishita, R. Sugano (Institute for Advanced Energy, Kyoto University), and B. D. Wirth (Department of Nuclear Engineering, University of California Berkeley) 213**

Extended Abstract

- 9.10 NUCLEATION AND GROWTH OF HELIUM-VACANCY CLUSTERS IN IRRADIATED METALS. PART II. A GROUPING METHOD FOR AN APPROXIMATE SOLUTION OF TWO DIMENSIONAL KINETIC EQUATIONS DESCRIBING EVOLUTION OF POINT DEFECT CLUSTERS TAKING INTO ACCOUNT BROWNIAN MOTION OF THE CLUSTERS—S. I. Golubov, R. E. Stoller, S. J. Zinkle (Oak Ridge National Laboratory) 214**

Nucleation, growth and coarsening of point defect clusters or secondary phase precipitates are of interest for many applications in solid-state physics. As an example, clusters nucleate and grow from point defects (PD) in solid under irradiation. In typical nucleation, growth and coarsening problems, a master equation (ME) is constructed that summarizes the large number of equations needed to describe the evolution process. When only the mobility of point defects and their reactions with the clusters are taken into account the ME takes the form of a differential equation known as the continuity equation in cluster size space. A new grouping method was developed by Golubov et al. for both the one-dimensional ME, which describes evolution of dislocation loops under irradiation or ageing, and the two-dimensional ME, which describes gas-assisted nucleation of voids or bubble formation in irradiated metals [1, 2]. However it has been shown that mobility of the clusters (e.g. He-vacancy) leading to coalescence, may play a key role in their evolution, particularly in the case of annealing of He implanted metals. The ME in the case becomes of the integro-differential type which complicates the numerical solution. The coalescence of clusters has been treated by different calculation methods (see e.g. [3-9]) but it has not been subjected to any specific grouping method of the type just described and this work intends to fill this gap. In the present work, the grouping method proposed by Golubov et al. [1] for the two-dimensional ME is generalized to take into account the coalescence of the clusters. An application of the method to the problem of helium bubble evolution which takes place during annealing of He implanted stainless steel is presented.

10.0 DOSIMETRY, DAMAGE PARAMETERS, AND ACTIVATION CALCULATIONS 232

10.1 IMPACT OF TRANSMUTATION ISSUES ON INTERPRETATION OF DATA OBTAINED FROM FAST REACTOR IRRADIATION EXPERIMENTS—L. R. Greenwood and F. A. Gamer (Pacific Northwest National Laboratory) 233

The subject of fission-fusion correlation is usually cast in terms of reactor-to-reactor differences, but recently the fusion community has become aware of the impact of differences within a given surrogate facility, especially in constant time experiments when different dose levels are attained in different positions of one reactor. For some materials, it is not safe to assume that in-reactor spectral variations are small and of no consequence.

11.0 MATERIALS ENGINEERING AND DESIGN REQUIREMENTS 240

No contributions

12.0 IRRADIATION FACILITIES AND TEST MATRICES 241

12.1 ASSEMBLY OF THE MFE-RB-17J EXPERIMENT—A. L. Qualls, K. R. Thoms, D. W. Heatherly, and R. G. Sitterson (Oak Ridge National Laboratory) 242

The 17J experiment is currently in the final stages of assembly in preparation for irradiation in a shielded RB position. The basic features of the design and assembly process are described. The specimen holders were loaded with specimens, filled with lithium, and then assembled into the experiment capsule, which will soon be connected to the control instrumentation.

12.2 ASSEMBLY OF THE US-JAPAN JP-26 EXPERIMENT AND START OF IRRADIATION IN THE HFIR—K. R. Thoms, D. W. Heatherly, S. H. Kim, R. G. Sitterson, R. E. Stoller (Oak Ridge National Laboratory), and H. Tanigawa (Japan Atomic Energy Research Institute, Tokai, Japan) 250

Specimen and capsule parts fabrication for JP-26 was completed. Loading of specimens into specimen holders and assembly of the capsule was completed. The experiment was installed in the target region of HFIR and irradiation began with cycle 398, starting December 11, 2003.