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He. These results suggest that an external oxide layer may be forming at the higher pressures. During exposures in 1 atm He, specimens form an external oxide layer and retain some tensile ductility after exposure. To confirm an earlier result, specimens with various exposure times (and thus oxygen levels) were annealed for 2000h at 700°C. The ductility of those specimens which had an external oxide before annealing was reduced to zero, illustrating that a surface oxide on V-4Cr-4Ti is a source of oxygen for further embrittlement.

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High purity single crystal and chemically vapor deposited silicon carbide has been amorphized under fast neutron irradiation. The gradual transition in physical properties from the as-amorphized state to a more relaxed amorphous state prior to crystallization is studied. For the three bulk properties studied: density, electrical resistivity, and thermal conductivity, large property changes occur upon annealing between the amorphization temperature and the point of crystallization. These physical property changes occur in the absence of crystallization and are attributed to short, and perhaps medium range ordering during annealing. It is demonstrated that the physical properties of amorphous SiC can vary greatly, and are likely a function of the irradiation state producing the amorphization. The initiation of crystallization as measured using bulk density and in-situ TEM is found to be ~ 875°C, though the kinetics of crystallization above this point are seen to depend on the technique used. It is speculated that in-situ TEM and other thin-film approaches to studying crystallization of amorphous SiC are faulted due to thin-film effects.

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The effects of neutron irradiation to ~7 dpa at 500 and 800°C on a high quality, three-dimensional balanced weave composite (FMI-222) is presented. Strength and dimensional stability for this system is compared to earlier work on this material, at lower dose, and contrasted with that of a well studied isotropic graphite (POCO AXF-5Q) irradiated at identical conditions. For both irradiation temperatures the composite strength in bending is substantially increased. Interestingly, while both irradiation temperatures cause contraction along the bend bar axis, the amount of contraction is greater for the higher temperature irradiation. Moreover, for the 500°C specimens nearly isotropic contraction leads to a corresponding decrease in volume, though an apparent large increase in volume occurs for the 800°C irradiated composite due to very anisotropic dimensional change. As the FMI-222 is a balanced-weave, isotropic composite, non-isotropic swelling behavior was unexpected, and is explained in

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Although the results of studies on reduced-activation ferritic/martensitic steels have demonstrated the excellent properties of 7-10Cr-2WVTa martensitic steels little information is available concerning the effect of chromium over the 5-10% range, where the steels transform to martensite when normalized or quenched. In this work, the effect of chromium, tungsten, and tantalum was investigated for the martensitic 5-9% Cr steels. The effect of boron was also investigated, because it has often been used in these types of steels to improve properties.

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Post-irradiation annealing experiments were conducted on neutron-irradiated pure copper (from 0.01 to 0.3 dpa at 100°C) to explore the possibility of mitigating the effects of radiation hardening on the strength and ductility. The post-irradiation annealing (PIA) condition of 300°C for 50 hours yielded mixed results and did not completely remove the effects of irradiation. However, there were changes brought about in the microstructure that led to the removal of the yield point phenomenon and restored some of the unirradiated ductility and work hardening. The true stress-strain curves have been calculated for both the as-irradiated and the PI annealed Cu and compared to that of the unirradiated

pure copper. The true stresses at which the as-irradiated samples begin to neck (based on the engineering stress-strain curves) are similar for the 0.1 to 0.3 dpa samples and are similar to that of the unirradiated copper. On the other hand, the as-irradiated 0.01 dpa condition and all of the post-annealed conditions begin necking at lower stresses, but at similar strains. The work hardening rate in the as-irradiated copper (0.1 to 0.3 dpa), when compared on Kocks-Mecking plots, shows that after a few percent strain, the macroscopic work hardening behavior mimics that of the unirradiated pure copper in the latter stages of work hardening. The PI annealing does little to restore the elongation compared to the as-irradiated state, yet lowers the yield strength, removes the yield point phenomenon and causes the material to begin necking at a lower stress. Compression testing is offered as a method to extend the mechanical properties testing to minimize the geometrical instabilities inherent in the tensile testing and explore the full range of work hardening up to failure.

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No contributions.

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The lower limit for void swelling of austenitic stainless steel was investigated using a flow restrictor component from the low-flux breeder zone of the BN-350 fast reactor in Kazakhstan. This component was constructed of annealed 12X18H10T, an alloy similar to AISI 321 which is used in Russian reactors for applications where AISI 304L would be used in comparable Western and Japanese reactors. Extensive sectioning to produce 114 separate specimens, followed by examination of the radiation-induced microstructure showed that void swelling in the range of temperatures and dpa rates of PWR interest occurs down to ~305°C. At 330°C the swelling reached ~1% at 20 dpa. Comparison of these data with other published results from Russian LWR reactors at <10 dpa confirms that the lowest temperature that stainless steels can begin swelling appears to be ~300°C. Since fusion and PWR spectra generate similar levels of hydrogen and helium, it is expected that these conclusions are also applicable to fusion devices operating at comparable dpa rates.

6.2 GENERATION AND RETENTION OF HELIUM AND HYDROGEN IN AUSTENITIC STEELS IRRADIATED IN A VARIETY OF LWR AND TEST REACTOR SPECTRAL ENVIRONMENTS - F. A. Garner, B. M. Oliver, L. R. Greenwood, D. J. Edwards and S.M. Bruemmer (Pacific Northwest National Laboratory), M. L. Grossbeck, (Oak Ridge National Laboratory) 127

In fission and fusion reactor environments stainless steels generate significant amounts of helium and hydrogen by transmutation. The primary sources of helium are boron and nickel, interacting with both fast and especially thermal neutrons. Hydrogen arises primarily from fast neutron reactions, but is also introduced into steels at often much higher levels by other environmental

processes. Although essentially all of the helium is retained in the steel, it is commonly assumed that most of the hydrogen is not retained. It now appears that under some circumstances, significant levels of hydrogen can be retained, especially when helium-nucleated cavities become a significant part of the microstructure. A variety of stainless steel specimens have been examined from various test reactors, PWRs and BWRs. These specimens were exposed to a wide range of neutron spectra with different thermal/fast neutron ratios. Pure nickel and pure iron have also been examined. It is shown that all major features of the retention of helium and hydrogen can be explained in terms of the composition, thermal/fast neutron ratio and the presence or absence of helium-nucleated cavities. In some cases, the hydrogen retention is very large and can exceed that generated by transmutation, with the additional hydrogen arising from either environmental sources and/or previously unidentified radioisotope sources that may come into operation at high neutron exposures.

6.3 NEUTRON-INDUCED MICROSTRUCTURAL EVOLUTION OF Fe-15Cr-16Ni ALLOYS AT ~ 400°C DURING NEUTRON IRRADIATION IN THE FFTF FAST REACTOR – T. Okita, T. Sato and N. Sekimura (University of Tokyo), F. A. Garner and L. R. Greenwood (Pacific Northwest National Laboratory), W. G. Wolfer (Lawrence Livermore National Laboratory), and Y. Isobe (Nuclear Fuel Industries Ltd., Japan) 148

An experiment conducted at ~400°C on simple model austenitic alloys (Fe-15Cr-16Ni and Fe-15Cr-16Ni-0.25Ti, both with and without 500 appm boron) irradiated in the FFTF fast reactor at seven different dpa rates clearly shows that lowering of the atomic displacement rate leads to a pronounced reduction in the transient regime of void swelling. While the steady state swelling rate (~1%/dpa) of these alloys is unaffected by changes in the dpa rate, the transient regime of swelling can vary from <1 to ~60 dpa when the dpa rate varies over more than two orders of magnitude. This range of dpa rates covers the full span of fusion, PWR and fast reactor rates. The origin of the flux sensitivity of swelling arises first in the evolution of the Frank dislocation loop population, its unfaulting, and the subsequent evolution of the dislocation network. There also appears to be some flux sensitivity to the void nucleation process. Most interestingly, the addition of titanium suppresses the void nucleation process somewhat, but does not alter the duration of the transient regime of swelling or its sensitivity to dpa rate. Side-by-side irradiation of boron-modified model alloys in this same experiment shows that higher helium generation rates homogenize the swelling somewhat, but do not significantly change its magnitude or flux sensitivity. The results of this study support the prediction that austenitic alloys irradiated at PWR-relevant displacement rates will most likely swell more than when irradiated at higher rates characteristic of fast reactors. Thus, the use of swelling data accumulated in fast reactors may possibly lead to an under-prediction of swelling in lower-flux PWRs and fusion devices.

6.4 **INVESTIGATION OF THE SYNERGISTIC INFLUENCE OF IRRADIATION TEMPERATURE AND ATOMIC DISPLACEMENT RATE ON THE MICROSTRUCTURAL EVOLUTION OF ION-IRRADIATED MODEL AUSTENITIC ALLOY Fe-15Cr-16Ni** – T. Okita, T. Iwai and N. Sekimura (Tokyo University), F. A. Garner (Pacific Northwest National Laboratory) 165

A comprehensive experimental investigation of microstructural evolution has been conducted on Fe-15Cr-16Ni irradiated with 4.0 MeV nickel ions in the High Fluence Irradiation Facility of the University of Tokyo. Irradiations proceeded to dose levels ranging from ~0.2 to ~26 dpa at temperatures of 300, 400 and 500°C at displacement rates of 1 x 10^{-4}, 4 x 10^{-4} and 1 x 10^{-3} dpa/sec. This experiment is one of two companion experiments directed toward the study of the dependence of void swelling on displacement rate. The other experiment proceeded at seven different but lower dpa rates in FFTF-MOTA at ~400°C. In both experiments the swelling was found at every irradiation condition studied to monotonically increase with decreases in dpa rate. The microstructural evolution under ion irradiation was found to be very sensitive to the displacement rate at all three temperatures. The earliest and most sensitive component of microstructure to both temperature and especially displacement rate was found to be the Frank loops. The second most sensitive component was found to be the void microstructure, which co-evolves with the loop and dislocation microstructure. These data support the prediction that void swelling will probably be higher in lower-flux fusion devices and PWRs at a given irradiation temperature when compared to irradiations conducted at higher dpa rates in fast reactors.

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Bulk specimens of two candidate compositions, CaO and AlN+0.04wt.%Y, for insulating coatings in a lithium-cooled fusion reactor have been exposed to lithium in 1000 h isothermal tests at 500°-800°C to determine the maximum temperature at which acceptable compatibility is likely. Because high mass losses for AlN were observed at 700°C, additional testing has focused on this temperature in order to examine what changes may slow the rate of dissolution. Changing from a vanadium alloy test capsule to a Mo capsule appears to have the largest effect at 700°C. For single crystal specimens of CaO, high mass losses have been observed after exposure at 600°C. In this case, changing to a Mo capsule or adding Ca or O to the lithium have not consistently showed a beneficial effect. For testing at 700°C, neither doping the Li with Ca or O altered the high mass losses. These results suggest that CaO may be limited to exposure temperatures of less than 600°C but that AlN may be able to operate above 600°C.

7.2 IN-SITU FORMATION OF CaO INSULATOR COATINGS ON VANADIUM ALLOYS - D. L. Smith, J-H. Park, and K. Natesan (Argonne National Laboratory) 178

A key issue for the self-cooled lithium blanket concept with a vanadium alloy structure is the development of an electrically insulating coating on the coolant channel walls to mitigate the magneto-hydrodynamic pressure drop in a high magnetic field. A systematic investigation of the thermodynamics and kinetics of oxygen and calcium interactions in the vanadium alloy/lithium system is being conducted to define the system parameters required for in-situ formation of a CaO coating on vanadium alloys. This paper presents results of theory and modeling as well as experimental results on the formation of CaO coatings on vanadium alloys after exposure at temperatures of 600 – 700°C to lithium with a small fraction of Ca added. Coatings of 10 - 30 μm with high electrical resistiviy have been formed.

7.3 DEVELOPMENT, CHARACTERIZATION, AND Li COMPATIBILITY OF AN ELECTRICALLY INSULATING CaO COATING ON V-4Cr-4Ti ALLOY - K. Natesan, M. Uz, D. L. Rink, and D. L. Smith (Argonne National Laboratory) 186

As part of the U.S. Department of Energy's Fusion Program, research is being conducted to develop an electrically insulating and liquid-Li-compatible CaO coating on V-4Cr-4Ti alloy. CaO has been identified as the primary coating candidate for this application, based on its high electrical resistivity and thermodynamic stability in Li at the temperatures of interest (400-700°C). This report deals with the development and characterization of a CaO coating deposited on V-4Cr-4Ti alloy specimens by a vapor transport process that uses double Ca-deposition/oxidation steps. The specimen surfaces were analyzed after each step during the process, and the surfaces, along with their cross-sections, were analyzed after exposure to liquid Li. The analysis and characterization were done by one or more techniques, including scanning electron microscopy, energy-dispersive X-ray analysis, X-ray diffraction, and measurement of electrical resistivity external to Li. Here, we present the new developments and results obtained since the last reporting period that ended on December 31, 2000.

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This report addresses the development and characterization of CaO and Y_2O_3 coatings that were applied on V-4Cr-4Ti alloy by an MOCVD process. Several coupon specimens were coated with CaO or Y_2O_3 by the process developed at ANL. The specimens were analyzed and the coatings were characterized before exposure to a liquid Li environment. The analysis and characterization used one or more of scanning electron microscopy (SEM), energy-dispersive X-ray analysis (EDX), and X-ray diffraction (XRD) techniques. Results are presented, with emphasis on microstructural analysis and electrical resistivity of the coatings. We plan to expose the coated specimens in liquid Li of varying chemistry at temperatures of 500-700°C.

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9.1 ATOMISTIC SIMULATION OF DISLOCATION-DEFECT INTERACTIONS IN CU – B. D. Wirth, V. V. Bulatov and T. Diaz de la Rubia (Lawrence Livermore National Laboratory) 206

The mechanisms of dislocation-defect interactions are of practical importance for developing quantitative structure-property relationships, mechanistic understanding of plastic flow localization and predictive models of mechanical behavior in metals under irradiation. In copper and other face centered cubic metals, high-energy particle irradiation produces hardening and shear localization. Post-irradiation microstructural examination in Cu reveals that irradiation has produced a high number density of nanometer sized stacking fault tetrahedra. Thus, the resultant irradiation hardening and shear localization is commonly attributed to the interaction between stacking fault tetrahedra and mobile dislocations, although the mechanism of this interaction is unknown. In this work, we present a comprehensive molecular dynamics simulation study that characterizes the interaction and fate of moving dislocations with stacking fault tetrahedra in Cu using an EAM interatomic potential. This work is intended to produce atomistic input into dislocation dynamics simulations of plastic flow localization in irradiated materials.

10. DOSIMETRY, DAMAGE PARAMETERS, AND ACTIVATION CALCULATIONS 213

10.1 NEUTRON DOSIMETRY AND DAMAGE CALCULATIONS FOR THE HFIR-RB-11J,12J IRRADIATIONS – L. R. Greenwood (Pacific Northwest National Laboratory) and C. A. Baldwin (Oak Ridge National Laboratory) 214

Neutron fluence measurements and radiation damage calculations are reported for the joint U.S. Japanese experiments RB-11J and RB-12J which were conducted in the removable beryllium (RB) position of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). These experiments employed a Eu_2O_3 thermal shield for the first time to reduce the thermal neutron fluence. The maximum total neutron fluence at midplane was 1.85×10^{22} n/cm^2 (9.5×10^{21} n/cm^2 above 0.1 MeV), resulting in about 5.5 dpa and 3.6 appm helium in type 316 stainless steel.

11. MATERIALS ENGINEERING AND DESIGN REQUIREMENTS 220

No contributions.

12. IRRADIATION FACILITIES AND TEST MATRICES 221

No contributions.