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1.1 PREPARATION AND IRRADIATION OF FUSION MOTA 2B -- M. L. Hamilton, R. M. Ermi (Pacific Northwest Laboratory) and A. M. Ermi (Westinghouse Hanford Company) 3

MOTA 2B was built to support the research and development efforts of three fusion materials program: the U.S. DOE Neutron Interactive Materials (NIMs) program. the Japanese university fusion materials program, and the IEA-sponsored BEATRIX-II experiment. The MOTA was inserted into the FFTF for irradiation beginning in cycle 12. Irradiation has been completed successfully through cycle 12B.2. Both the specimen loading and the history of cycle 12 to date are documented here.

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A complete description and details of the design, construction, and installation of capsules JP-9 through JP-16 has been previously reported. The capsules were installed in the High Flux Isotope Reactor (HRR) target July 20, 1990, for irradiation beginning with HFIR fuel cycle 289. The capsules were removed and stored in the reactor pool during HRR cycle 292 (11/25/90-12/10/90) to provide room for required isotope production. They were reinstalled for HRR cycle 293 for continued irradiation. Of these eight target capsules, JP-10, -11, -13, and -16 completed their scheduled number of cycles (11) and were removed from the reactor in September 1991. These four experiments were successfully disassembled in January 1992.

Three new capsules in this series, JP-20, -21, and -22. are presently being designed. These capsules were added to the program in order to complete the experimental matrix included in the JP-9 through -16 capsules. The new capsules will contain transmission electron microscope (TEM) disks and SS-3 flat tensile specimens at 300 lo 600°C and will achieve doses of 8, 18, and 40 dpa. respectively.

1.3 FABRICATION AND IRRADIATION OF HFIR-MFE-JP-17, -18, AND -19 TARGET IRRADIATION CAPSULES -- A. W. Longest, D. W. Heatherly, K. R. Thoms (Oak Ridge National Laboratory) and J. E. Corum (Midwest Technical, Inc.) 24

Fabrication and irradiation of three new uninstrumented HRR target capsules for testing 12.5-mmdiam stainless steel fracture toughness Specimens to a damage level of approximately 3 displacements per atom (dpa) at temperatures of 60 to 125 and 250 to 300°C were completed this report period. Two low-temperature capsules of identical design, designated HRR-MFE-JP-18 and HFIR-MFE-JP-19, each contained 32 fracture toughness specimens directly cooled by reactor cooling water. Irradiation of these two capsules was completed on October 19, 1991. A single helium-filled elevated-temperature capsule, designated HFIR-MFE-JP-17, contained a stack of 86 fracture toughness specimens. Additional neutronic calculations were required for this experiment to insure that it would not cause unacceptable neutron flux shifting and hot spots in the HRR fuel regions. Irradiation of this capsule was completed on February 27, 1992. Included in each capsule were companion transmission electron microscopy (TEM) and SS-3 tensile specimens for correlation of microstructural. tensile. and fracture toughness properties.

1.4 FABRICATION AND OPERATION OF HFIR-MFE RB* SPECTRALLY TAILORED IRRADIATION CAPSULES - A. W. Longest, D. W. Heatherly, E. D. Clemmer (Oak Ridge National Laboratory), and J. E. Corum (Midwest Technical, Inc.) 29

Fabrication and operation of four HFIR-MFE RB* capsules (60,200,330. and 400°C) to accommodate MFE specimens previously irradiated in spectrally tailored experiments in the ORR are proceeding satisfactorily. With the exception of the 60°C capsule, where the lest specimens are in direct contact with the reactor cooling water. specimen temperatures (monitored by 21 thermocouples) are controlled by varying the thermal conductance of a thin gas gap region between the specimen holder and containment tube.

Irradiation of the 60 and 330°C capsules was started on July 17, 1990. As of March 31.1992, these two capsules had completed 17 cycles of their planned 22-cycle (formerly 28-cycle) irradiation to a damage level of approximately 17.5 displacements per atom (dpa). Assembly of the 200 and 400°C capsules is scheduled for completion in August, 1992; operation of these two capsules will follow the first two (60 and 330°C).

1.5 STATUS OF AUTOMATED TENSILE MACHINE -- M. Satou (Tohoku University), M. L. Hamilton (Pacific Northwest Laboratory) S. Salo and A Kohyama (University of Tokyo) 31

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Special futures and test methods have been developed for testing small disk compact specimens (12.5 mm diam by 4.6 mm thick). Both unloading compliance and potential drop methods have been used to monitor crack extension during the I-inlcgral resistance (I-R) curve testing. Provisions have been made Io allow the necessary probes and instrumentation to he installed remotely using manipulators for testing of irradiated specimens in a hot cell. Laboratory trials showed that both unloading compliance and potential drop gave useful results. Both techniques gave similar data. and predicted the final crack extension within allowable limits. The results from the small disk compact specimens were similar to results from conventional compact specimens 12.7 mm thick. However. the slopes of the I-R curves from the larger specimens were lower, suggesting that the smaller disk compact specimens may have lost some constraint due to their size. The testing show that it should be possible to generate useful J-R curve fracture toughness data from the small disk compact specimens.

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Neutron tluence and spectral measurements and radiation damage calculations are reported for the MOTA-2A experiment in the Fast Flux Test Facility (FFTF). The irradiation was conducted from January 4, 1990, to March 19, 1991, for a total exposure of 299.7 EFPD. The maximum fluence was 15.8 $\times 10^{22}$ n/cm^2, 10.0 $\times 10^{22}$ above 0.1 MeV which produced 43.0 dpa in iron. This MOTA assembly contained the most comprehensive dosimetry to date with thirteen complete spectral measuremenu and twenty additional flux gradient measuremenu.

2.2 CALCULATION OF DISPLACEMENT LEVELS FOR PURE ELEMENTS AND MOST MULTI-COMPONENT ALLOYS IRRADIATED IN FFTF MOTA-1F -- F. A. Garner and L. R. Greenwood (Pacific Northwest Laboratory) and A. M. Ermi (Westinghouse Hanford Company) 54

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Radiation damage will be a major key point in the design of the many duplex components in fusion reactors. There is a substantial amount of available data showing that stress plays a major role in the onset, and possibly the rate. of void growth in austenitic stainless steels. There is also strong support models which predict a coupling of swelling and creep through the stress environment. A parametric study for evidence to stress-enhanced swelling and its connection to creep is conducted for a typical fusion power demonstration reactor. Since microstructural changes are known to affect elastic moduli. the impact of stress enhanccd swelling on these moduli are also evaluated.

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5.1 TENSILE PROPERTY CHANGES OF METALS IRRADIATED TO LOW DOSES WITH FISSION, FUSION AND SPALLATION NEUTRONS -- H. L. Heinisch and M. L. Hamilton (Pacific Northwest Laboratory), W. F. Sommer (Los Alamos National Laboratory) and P. D. Ferguson (University of Missouri-Rolla) 75

Radiation effects due to low doses of spallation neutrons are compared directly to those produced by fission and fusion neutrons. Yield stress changes of pure Cu, alumina-dispersion-strengthened Cu and AISI 316 stainless steel irradiated at 36 to 55°C in the Los Alamos Spallation Radiation Effects Facility (LASREF) are compared with earlier results of irradiations at 90°C using 14 MeV D-T fusion neutrons at the Rotating Target Neutron Source and fission reactor neutrons in the Omega West Reactor. At doses up to 0.04 displacements per atom (dpa), the yield stress changes due to the three quite different neutron spectra correlate well on the basis of dpa in the stainless steel and the Cu alloy. However. in pure Cu. the measured yield stress changes due to spallation neutrons were anomalously small and should be verified by additional irradiations. With the exception of pure Cu, the low dose, low-temperature experiments reveal no fundamental differences in radiation hardening by fission, fusion, or spallation neutrons when compared on the basis of dpa.

5.2 A COMPARISON OF MICROSTRUCTURES IN COPPER IRRADIATED WITH FISSION, FUSION AND SPALLATION NEUTRONS -- T. Muroga (Kyushu University), H. L. Heinisch (Pacific Northwest Laboratory) W. F. Sommer (Los Alamos National Laboratory) and P. D. Ferguson (University of Missouri-Rolla) 81

The microstructures of pure copper irradiated to low doses at 36-90°C with spallation neutrons, fusion neutrons and fission neutrons are compared. The defect cluster densities for the spallation and fusion neutrons are very similar when compared on the basis of displacements per atom (dpa). In both cases, the density increases in proportion to the square root of the dpa. The difference in defect density between fusion neutrons and fission neutrons corresponds with differences observed in data on yield stress changes.

5.3 MOLECULAR DYNAMICS AND BINARY COLLISION MODELING OF THE PRIMARY DAMAGE STATE OF COLLISION CASCADES -- H. L. Heinisch (Pacific Northwest Laboratory) and B. N. Singh (Risd National Laboratory) 86

Quantitative information on defect production in cascades in copper obtained from recent molecular dynamics (MD) simulations is compared to defect production information determined earlier with a model based on the binary collision approximation (BCA). The total numbers of residual defects. the fractions of them that are mobile, and the sizes of immobile clusters compare favorably. especially when the termination conditions of the two simulations are taken into account. A strategy is laid out for integrating the details of the cascade quenching phase determined by MD into a BCA-based model that is practical for simulating much higher energies and longer times than MD alone can achieve. The extraction of collisional phase information from MD simulations and the correspondence of MD and BCA versions of the collisional phase is demonstrated at low energy.

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Radiation-induced segregation (RIS) has been measured by analytical electron microscopy in a number of neutron-irradiated stainless steels stabilized with nickel or manganese. RIS to high-angle grain boundaries, cavities. and dislocation loops was observed. Similar enrichment of nickel, silicon, and, in certain cases, phosphorus and depletion of chromium, iron. molybdenum, and manganese occurred at all three of these defect sinks. The origin of matrix composition fluctuations is discussed and an alternate mechanism. based on RIS to the initial dislocation loop microstructure, is suggested.

5.5 A MICROSTRUCTURAL EXAMINATION OF THE MECHANICAL RESPONSE OF FeCrNi ALLOYS IRRADIATED AT 365°C USING ISOTOPIC TAILORING TO PRODUCE DIFFERENT He/dpa RATIOS -- J. F. Stubbins (University of Illinois) and F. A. Garner (Pacific Northwest Laboratory) 98

A series of model Fe-15Cr-xNi-yP alloys with x = 25 or 45 and y = 0.001 or 0.04 wt % were examined following irradiation to 10.3 dpa at 365°C. These alloys were irradiated both doped and undoped with the ^{59}Ni isotope to produce two levels of helium generation during irradiation. This latter variable produced average He/dpa values of approximately 1 and 16 appm He/dpa in the undoped and doped cases, respectively. Microscopy examination indicates that the major influence of a high He/dpa ratio is to induce a higher void number density and a lower average void size than in identical alloy conditions with low He/dpa. The total swelling is not substantially affected, however. Compositional and thermal mechanical treatment variables were found to influence the swelling behavior as much as the helium production rate. However, in all cases the swelling was low. and irradiation-induced microstructures were similar regardless of starting condition. This convergence in microstructure is reflected in the mechanical properties where similar room temperature tensile properties were measured on irradiated specimens regardless of material starting condition.

5.6 MICROSTRUCTURES OF NEUTRON-IRRADIATED Fe-12Cr-XMn (X=15-30) TERNARY ALLOYS -- K. Miyahara (Nagoya University), F. A. Garner (Pacific Northwest Laboratory), and Y. Hosoi (Nagoya University) 104

The Fe-Cr-Mn alloy system is being studied as an alternative to the Fe-Cr-Ni system because of the need to reduce long-term radioactivation in fusion-power devices. In this study, four Fe-12Cr-XMn (X = 15, 20, 25, 30 wt %) alloys were irradiated in the Fast Flux Test Facility to 20 dpa at 643 K and 40 dpa at 679. 793, and 873 K to investigate the influence of manganese content on void swelling and phase stability. The results confirm and expand the results of earlier studies that indicate that the Fe-Cr-Mn system is relatively unstable compared lo that of the Fe-Cr-Ni system. with alpha and sigma phases forming as a consequence of thermal aging or high-temperature irradiation.

5.7 MICROSTRUCTURAL EVOLUTION OF NEUTRON-IRRADIATED Ni-Si AND Ni-AI ALLOYS -- H. Takahashi (Hokkaido University) and F. A. Garner (Pacific Northwest Laboratory) 111

Additions of silicon and aluminum suppress the neutron-induced swelling of pure nickel but to different degrees. Silicon is much more effective initially when compared to aluminum on a per-atom basis, but silicon exhibits a non-monotonic influence on swelling with increasing concentration. Silicon tends to segregate toward grain boundaries while aluminum segregates away from these boundaries. Whereas the formation of the Ni_3Si phase is frequently observed in charged particle irradiation experiments conduted at much higher displacement rates. it did not occur during neutron irradiation in this study. Precipitation also did not occur in Ni-5Al during neutron irradiation. nor has it been reported to occur during ion irradiation.

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6.1.1 ANALYSIS OF STRESS-INDUCED BURGERS VECTOR ANISOTROPY IN AN IRRADIATED FERRITIC-MARTENSITIC STEEL: JFMS -- D. S. Gelles (Pacific Northwest Laboratory), A. Kimura (Muroran Institute of Technology). A. Kohyama (University of Tokyo), and R. J. Puigh (Westinghouse Hanford Company) 123

A procedure for determining the Burgers vector anisotropy in irradiated ferritic steels is described. allowing identification of all a<100> and all (a/2)<111> dislocations in a region of interest. The necessary micrographs to apply the procedure have been taken of two pressurized tube specimens of JFMS. a duplex ferritic steel following irradiation in FFTF/MOTA at 407°C to 7.5 X 10^{22} n/cm^2 (E > 0.1 MeV) or 37.5 dpa. Preliminary analysis of one set of micrographs indicates large anisotropy in Burgers vector populations for an 86 MPa stressed condition.

6.1.2 FERRITIC/MARTENSITIC STEELS: PROMISES AND PROBLEMS -- R. L. Klueh (Oak Ridge National Laboratory), K. Ehrlich (Kernforschungszentrum Karlsruhe) and F. Abe (Tsukuba Laboratories) 131

Ferriticlmartensitic steels are candidate structural materials for fusion reactors because of their higher swelling resistance, higher thermal conductivity, lower thermal expansion, and better liquid-metal compatibility than austenitic steels. Irradiation effects will ultimately determine the applicability of these steels, and the effects of irradiation on microstructure and swelling and on the tensile, fatigue. and impact properties of the ferriticimartensiticsteels are discussed. Most irradiation studies have been carried out in fast reactors. where little transmutation helium forms. Helium has been show to enhance swelling and affect tensile and fracture behavior. making helium a critical issue, since high helium concentrations will be generated in conjunction with displacement damage in a fusion reactor. These issues are reviewed to evaluate the status of ferriticlmartensitic steels and to assess the research required to insure that-such steels are viable candidates for fusion applications.

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Reanalysis of an earlier data set derived from irradiation of long creep tubes in EBR-II at 550°C has shown that the creep-swelling coupling coefficient is relatively independent of temperature at approximately 0.6×10^{-2} MPa^{-1}, but falls with increases in the swelling rate especially at high stress levels. The action of stress-affected swelling and carbide precipitation exert different influences on the derivation of this coefficient.

6.2.2 ANALYSIS OF CREEP DATA FROM MOTA IRRADIATION OF 20% COLD WORKED 316 STAINLESS STEEL -- F. A. Garner (Pacific Northwest Laboratory), M. B. Toloczko (University of California-SB), and R. J. Puigh (Westinghouse Hanford Company) 148

Analysis of creep data for 20% cold-worked 316 stainless steel irradiated in FFTF/MOTA demonstrates that the creep-swelling coupling coefficient is not a strong function of temperature and can be assumed to be approximately 0.6 X 10^{-2} MPa^{-1} in the range 400-600°C. It appears, however, that the creep compliance B_o is a moderately strong function of displacement rate, especially at lower irradiation temperatures. It also appears to be a function of temperature and alloy composition. The latter dependency arises primarily because derived values of B_o unavoidably incorporate precipitation-related strains that cannot be easily separated from contributions arising from true creep. It has also been found that at approximately 550-600°C there is an upper limit on the total diametral strain rate at 0.33%/dpa. In contrast to the conclusion of an earlier experiment, this limitation does not arise initially from the total disappearance of creep, however. The creep rate first increases with the onset of swelling and then diminishes as the swelling rate increases, disappearing only when the swelling rate reaches its steady state value.

6.2.3 VOID SWELLING RESISTANCE OF PHOSPHORUS-MODIFIED AUSTENITIC STAINLESS STEELS DURING HFIR IRRADIATION TO 57 dpa AT 300 TO 500°C -- P. J. Maziasz (Oak Ridge National Laboratory) 163

The austenitic stainless steel prime candidate alloy (PCA) is a titanium-modified alloy (14Cr-16Ni) developed for void swelling resistance at 400 to 600°C by the U.S. Fusion Reactor Materials (FRM) program. During irradiation in the High Flux Isotope Reactor (20-70 appm Heidpa) at 300-600°C, the PCA (and the closely related D9-type alloys) has been shown to be highly susceptible to void swelling at 500°C after 34-57 dpa. Generally such alloys show good void swelling resistance to higher doses during irradiation in fast-breeder reactors (FBRs, 0.5-1 appm Heidpa) in the 20-25% cold-worked condition. New multiply stabilized. phosphorus-modified PCA alloys have been dcvcloped, with specific amounts and combinations of minor alloy elements added for better MC formation and stability characteristics during irradiation. After HFIR irradiation at 300-500°C to 34-57 dpa, the multiply stabilized, phosphorus-modified PCA and phosphorus-modified D9 alloy showed better swelling resistance (by density-change measurements) than similar alloys without phosphorus, particularly at 500°C. Microstructural studies after 34 dpa at 500°C showed that void swelling resistance in the multiply stabilized, phosphorus-modified PCA alloys was directly related to the formation of ultrafine dispersions of MC precipitates during HFIR irradiation.

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6.3.1 NEUTRON IRRADIATION DAMAGE OF A STRESS RELIEVED TZM ALLOY -- K. Abe, T. Masuyama, and M. Satou (Tohoku University), and M. L. Hamilton (Pacific Northwest Laboratory) 177

Disk specimens of the molybdenum alloy TZM that had been stress relieved at 1199 K (926°C) for 0.9 ks (15 min) were irradiated in the FFTF/MOTA 1F at 679, 793, and 873 K (406, 520, and 600°C) to a fast fluence of approximately 9.6 X 10^{22} n/cm^2. Microstructures were observed in a transmission electron microscope (TEM). Dislocation structures consisted of isolated loops, aggregated loops (rafts) and elongated dislocations. The size of the loops increased with the irradiation temperature. Void swelling was about 1 and 2% at 793 and 873 K (520 and 600°C), respectively. A void lattice was developed in the body centered cubic (bcc) structure with a spacing of 26 to 28 nm. The fine grain size (0.5 -2 μm) was retained following high-temperature irradiation, indicating that the stress relief heat treatment may extend the material's resistance to radiation damage up to high tluence levels. Microhardness measurements indicated that irradiation hardening increased with irradiation temperature. The relationship between the microstructure and the observed hardening was determined.

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6.4.1 DISPERSOID STABILITY IN A $Cu\text{-}Al_2O_3$ ALLOY UNDER ENERGETIC CASCADE DAMAGE CONDITIONS -- S. J. Zinkle (Oak Ridge National Laboratory), A. Horsewell and B. N. Singh (Risø National Laboratory), and W. F. Sommer (Los Alamos National Laboratoty) 187

A commercial dispersion-slrcngthened $Cu\text{-}Al_2O_3$ alloy was irradiated with 750-MeV protons at 470 K to a damage level of about 2 displacements per atom. The density and size distribution of the Al_2O_3 particles was measured in nonirradiated and irradiated specimens using transmission electron microscopy. The mean primary knock-on atom (PKA) energy for 750-MeV protons in copper is 2.5 MeV, which is about 10 and 100 times higher than the average PKA energies in copper for fusion and fission neutrons. respectively. The irradiation caused only a slight decrease in the mean Al_2O_3 size, from 10.5 nm to 8.3 nm and a slight decrease in the particle density from 4 to $3 \times 10^{22}/m^3$. These results suggest that $Cu\text{-}Al_2O_3$ alloys. which are resistant Io radiation-induced microstructural changes during fission neutron irradiation, may also be microstructurally stable in a fusion neutron environment.

6.4.2 FISSION NEUTRON IRRADIATION OF COPPER CONTAINING IMPLANTED AND TRANSMUTATION PRODUCED HELIUM -- B. N. Singh. A. Horsewell and M. Eldrup (Risø National Laboratory) and F. A. Garner (Pacific Northwest Laboratory) 192

High purity copper containing approximately 100 appm helium was produced in two ways. In the first. helium was implanted by cyclotron at Harwell at 323 K. In the sewnd method, helium was produced as a transmutation product in 800 MeV proton irradiation at Los Alamos. also at 323 K. The distributions of helium prior to fission neutron irradiation were determined by a combination of transmission electron microscopy (TEM) and positron annihilation techniques (PAT). These specimens. together with pure copper. were then irradiated with fission neutrons in a single capsule in Fast Flux Test Facility (FFTF) at approximately 686 ±5 K to a dose level of approximately 48 dpa (7.7×10^{26} n.m.$^{-2}$; E > 0.1 MeV). Investigation of the void and dislocation microstructures in the three specimens by TEM showed large differences between the specimens in void size and swelling. The observed differences as well as the effect of the presence of other transmutation-produced impurity atoms in the 800 MeV proton irradiated copper will be discussed,

6.4.3 RESPONSE OF SOLUTE AND PRECIPITATION STRENGTHENED COPPER ALLOYS AT HIGH NEUTRON EXPOSURE -- F. A. Garner and M. L. Hamilton (Pacific Northwest Laboratory), T. Shikama (Tohoku University), and D. J. Edwards and J. W. Newkirk (University of Missouri) 198

A variety of solute and precipitation strengthened copper base alloys have been irradiated to neutron-induced displacement levels of 34 to 150 dpa at 415°C and 32 dpa at 529°C in the Fast Flux Test Facility to assess their potential for high heat flux applications in fusion reactors. Several MZC-type alloys appear to offer the most promise for further study. For low fluence applications CuBeNi and spinodally strengthened CuNiTi alloys may also he suitable. Although Cu-2Be resists swelling, it is not recommended for fusion reactor applications because of its low conductivity.

6.4.4 TENSILE AND FRACTURE BEHAVIOR OF Cu-5Ni AND VARIOUS PRECIPITATION-STRENGTHENED COPPER ALLOYS AFTER HIGH FLUENCE IRRADIATION -- T. Shikama (Tohoku University), F. A. Garner and M. L. Hamilton (Pacific Northwest Laboratory) and K. R. Anderson (University of Illinois) 205

Tensile tests and fractography were used to examine the radiation-induced evolution of Cu-5Ni and three commercially prepared precipitation-strengthened alloys after irradiation at 685 and 800 K (411 and 527°C). The addition of nickel to copper resulted in increased ductility prior to irradiation but decreased relative ductility after irradiation. The difference in behavior following irradiation appears to arise from the influence of nickel on void swelling. The precipitation-strengthened alloys in general exhibited a drop in yield strength after irradiation and an associated increase in ductility. The major effect in these alloys appeared to be related to overaging of the precipitates.

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6.5.1 ASSESSMENT OF H-INDUCED CRACK GROWTH OF TYPE 316 SS AT ITER CONDITIONS -- R. H. Jones (Pacific Northwest Laboratory) 213

The results of this analysis suggest that H-induced crack growth of Type 316 SS is a definite concern for ITER first-wall applications. It was estimated that crack velocities as high as 2×10^{-6} m/s could result in Type 316 SS irradiated at 100°C to 10 dpa, while the threshold, K_{th}, for crack growth could be as low as 28 to 37 MPa$\sqrt{m}$ for hydrogen generated by (n,p) reactions. This crack velocity would propagate through a 4-mm-thick first wall in 33 min, and therefore these conclusions should be evaluated experimentally. Also, a K_{th} of 28 MPa$\sqrt{m}$ requires that the applied and residual stresses in the structure be limited to 500 MPa for a I-mm-long crack to avoid H-induced crack growth. Higher stress limits are possible for shorter cracks. These predictions are based on the assumption that all the H generated by (n,p) reactions would be contained within the material; however, some loss of H to the vacuum or water side of the first wall is expected. The extent of this loss will depend on engineering considerations such as coatings, etc.

6.5.2 AQUEOUS STRESS CORROSION OF CANDIDATE AUSTENITIC STEELS FOR ITER STRUCTURAL APPLICATIONS -- A. B. Hull, P. R. Luebbers, M. R. Fox, W. K. Soppet, and T. F. Kassner (Argonne National Laboratory) 22s

Susceptibility of Types 316NG and sensitized 304 stainless steels (SS) to SCC was investigated at temperatures of 60-289°C in slow-strain-rate-tensile (SSRT) tests in oxygenated water that simulates important parameters anticipated in first-wall/blanket systems. Type 316NG SS exhibits good resistance to SCC under crevice and noncrevice conditions at temperatures <150°C in nominal ITER coolant chemistry that does not contain any short-lived radical species from radiolysis of water. Initial SSRT tests have been conducted on weldment specimens of Type 316NG SS with matching filler metal under crevice and noncrevice conditions in oxygenated water at 95°C. These specimens fractured in the base metal rather than in the weld or heat-affected zones.

6.5.3 CHARACTERIZATION OF RADIATION-INDUCED SEGREGATION IN AUSTENITIC STAINLESS STEELS USING ANALYTICAL ELECTRON MICROSCOPY AND SCANNING AUGER MICROPROBE TECHNIQUES -- G.E.C. Bell, E. A. Kenik and L. Heatherly, Jr. (Oak Ridge National Laboratory) 236

In this work, we characterized radiation-induced grain-boundary segregation of commercial-purity (CP) and high-purity (HP) Type 304 SS irradiated in a fission power reactor at 288°C to 2×10^{11} n/cm^2 (E > 1 MeV) or about 1 dpa using both high-resolution AEM and Auger electron spectroscopy (AES). Both AEM and AES showed strong nickel and silicon enrichment along with moderate chromium depletion at grain boundaries in the CP material. Chromium and nickel segregation, measured both by AEM and AES, were less in the HP material. No statistically significant minor element segregation was measurable with either AEM or AES. Good agreement between AEM segregation profiles measured perpendicular to grain boundaries and AES sputter-depth profile results was obtained.

6.5.4 ELECTROCHEMICAL AND MICROSTRUCTURAL CHARACTERIZATION OF AN AUSTENITIC STAINLESS STEEL IRRADIATED BY HEAVY IONS ABOVE 600°C -- G.E.C. Bell, E. A. Kenik (Oak Ridge National Laboratory) and T. Inazumi (Japan Atomic Energy Research Institute) 244

The electrochemical behavior and microstructural evolution of a solution-annealed. heavy-ion-irradiated. austenitic stainless steel (SS) have been investigated at temperatures above 600°C at doses of 1, 10, and 70 displacements per atom with concurrent helium injection of 0.2 at. ppm He per dpa. Changes in the electrochemical properties were measured by the single-loop electrochemical potentiokinetic reactivation (SL-EPR) technique using transmission electron microscopy disk specimens. Microchemical analysis has also been performed on some duplicate specimens in an AEM. Electrochemical results showed that the reactivation charge per unit etched area and Flade potential increased with increasing dose. In the case of the 70 dpa specimen, complete passivation of the specimen could not be obtained. AEM microanalysis showed changes in composition at both grain boundaries and faulted dislocation loops of chromium, iron, nickel, and silicon due to RIS.

6.5.5 EFFECTS OF COMPOSITIONAL MODIFICATIONS ON THE SENSITIZATION BEHAVIOR OF Fe-Cr-Mn STEELS -- G. L. Edgemon (Georgia Institute of Technology), P. F. Tortorelli and G.E.C. Bell (Oak Ridge National Laboratory) 250

Fe-Cr-Mn steels may possibly be used in conjunction with aqueous blankets or coolants in a fusion device. Therefore, standard chemical immersion (modified Strauss) tests were conducted to characterize the effects of compositional modifications on the thermal sensitization behavior of these steels. A good correlation among weight losses, intergranular (IG) corrosion, and cracking was found. The most effective means of decreasing their susceptibility was through reduction of the carbon concentration of these steels to 0.1%, but the sensitization resistance of Fe-Cr-Mn-0.1C compositions was still inferior to type 304L and other similar stainless steels (SS). Alloying additions that form stable carbides did not have a very significant influence on the sensitization behavior.

6.5.6 LITHIUM PURIFICATION AT TEMPERATURES BELOW 500°C -- G.E.C. Bell (Oak Ridge National Laboratory) 256

A method for removal of nitrogen from liquid lithium has been investigated that allows nitrogen purification to be done at temperatures below 500°C. The method utilizes the formation of an insoluble, ternary compound on chromium-plated stainless steel surfaces lo remove nitrogen. Capsule tests established the initial feasibility of the method and determined optimum warm-trapping parameters for a large-scale batch test. Favorable results from capsule tests were obtained with processing temperatures of 450°C and chromium surface area-to-lithium volume ratios greater than 100 m^{-1}. Nitrogen levels as low as 37 weight parts per million were achieved in less than 30 h at 450°C by using a staged. large-scale batch process. When used in conjunction with conventional cold trapping, the method allows lithium purification to be done in a single, inexpensive vessel with a higher degree of safety, as compared to higher temperature methods (e.g., hot gettering).

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7.1 IN-SITU TRITIUM RECOVERY FROM Li_2O IRRADIATED IN FAST NEUTRON FLUX - BEATRIX-II TEMPERATURE CHANGE SPECIMEN -- O. D. Slagle and G. W. Hollenherg (Pacific Northwest Laboratory), T. Kurasawa (Japan Atomic Energy Research Institute). and R. A. Verrall (AECL Research) 265

The BEATRIX-II irradiation experiment is an in-situ tritium release experiment lo evaluate the stability and tritium release characteristics of Li_2O under fast neutron irradiation to extended burnups. A thin annular ring specimen capable of temperature changes was irradiated in Phase I of the experiment to a lithium burnup of 5%. The primary emphasis of the test plan was to determine the effect and interrelationship of gas composition and temperalure on the tritium recovery from Li_2O. Temperature changes in the range from 500 to 650°C resulted in a decreasing tritium inventory with increasing temperature and a series of specific temperature changes were carried out at intervals throughout the experiment to characterize the effect of burnup. Decreasing the amount of hydrogen in the sweep gas resulted in an increase in the tritium inventory in the Li_2O specimen. The tritium recovery during startup and shutdown was observed to be strongly influenced by the composition of the sweep gas.

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8.1 RADIATION-INDUCED CHANGES IN THE PHYSICAL PROPERTIES OF CERAMIC MATERIALS -- S. J. Zinkle (Oak Ridge National Laboratory) and E. R. Hodgson (EURATOM/CIEMAT Fusion Association. Madrid, Spain) 273

The results lrom recent studies on radiation-induced changes in the electrical conductivity, loss tangent. and thermal conductivity of ceramics are reviewed, with particular emphasis on in situ data obtained during irradiation. The in situ measurements have found that the physical property degradation during irradiation is generally much more severe than indicated by post-irradiation mcasurcments. In particular, permanent degradation in the clectrical resistivity may occur after irradiation to damage levels of $<10^{-3}$ displacements per atom (dpa) for temperatures near 450°C. This radiation-induced electrical degradation (RIED) is observed only if an electric field is applied during the irradiation.

8.2 MATERIALS ISSUES IN DIAGNOSTIC SYSTEMS FOR BPX AND ITER -- F. W. Clinard, Jr. and E. H. Farnum (Los Alamos National Laboratory) D. L. Griscom (Naval Research Laboratory), R. F. Mattas (Argonne National Laboratory), S. S. Medley (Princeton Plasma Physics Laboratory), F. W. Wiffen (U.S. Department of Energy), S. S. Wojtowicz (General Atomics), K. M. Young (Princeton Plasma Physics Lahoratory), and S. J. Zinkle (Oak Ridge National Laboratory) 283

Anticipated radiation fluxes in D-T burning machines such as BPX and ITER arc intense, and accumulated damage levels are moderate to severe. Thus materials used in diagnostic systems may suffer significant degradation of electrical. optical, and structural properties, either lrom transient or permanent damage effects. Of particular concern arc windows, optical fibers, reflectors, and insulators. However, materials design and modification based on results from an appropriate irradiation testing program, when combined with design optimization for location. shielding. and ease of replacement, should point the way to development of acceptable materials systems.

8.3 OBSERVATION OF RADIATION-INDUCED CHANGES IN DIELECTRIC PROPERTIES IN CERAMIC INSULATORS -- R. E. Stoller, R. H. Goulding, and S. J. Zinkle, and D. H. Rasmassen (Oak Ridge National Laboratory) *288*

Several experiments have been completed in which the dielectric properties of ceramic materials were measured during ionizing and displacive irradiation. The experimental matrix included single crystal sapphire, commercial-grade polycrystalline alumina. magnesium-aluminate spinel, silicon nitride, and aluminum nitride. These materials are candidates for use in a number of fusion reactor components. The measurements were made at 100 MHz using a capacitively loaded resonant cavity. Essentially. no effect of ionizing radiation alone was observed for damage rates up to about 5 x I d Gy/h. However. pulsed fission reactor irradiation led to dramatic increases in the loss tangent. The fractional contributions of the ionizing and displacive fields to the total change observed during the neutron irradiations could not be unambiguously determined. The results of these experiments indicate that the use of post-irradiation measurements leads to an underestimate of the radiation-induced dielectric degradation that occurs in the radiation field.

8.4 MEASUREMENT OF ELECTRICAL AND OPTICAL PROPERTIES OF DIELECTRIC MATERIALS DURING NEUTRON IRRADIATION -- E. H. Farnum, F. W. Chard, Jr., J. C. Kennedy III, W. F. Sommer and W. P. Unruh (Los Alamos National Laboratory) **296**

An in situ neutron irradiation experiment will he carried out at Los Alamos National Laboratory during the summer of 1992, to assess behavior of electrical insulators and optical fibers during neutron irradiation. Testing will be carried out at the Los Alamos Spallation Radiation Effect Facility (LASREF). Experiments include measurement of ac and dc electrical properties of Al_2O_3 at 300, 400, and 500°C. with applied electric fields up to 2000 V/cm. Additional measurements will be carried out on MgO-insulated instrumentation cables. Tests on optical fibers include attenuation, fluorescence, and reflectance measurements. at ambient temperature and moderate to intense neutron fluxes. Results will be applied to identification and improvement of candidate materials for ITER and other fusion devices.

8.5 THE EFFECT OF PHONON SCATTERING BY ALUMINUM PRECIPITATES AND VOIDS ON THE THERMAL CONDUCTIVITY OF Al_2O_3 -- D. P. White (Oak Ridge National Laboratory) **298**

Recent calculations on the effect of vacancies and conduction band electrons on the thermal conductivity of Al_2O_3 have shown that large decreases in the conductivity may be expected due to the scattering of phonons by vacancies in material exposed to displacive radiation. This work has been extended to include the effects of aluminum precipitates and voids on the lattice thermal conductivity. The change in the thermal conductivity due to these extended defects is presented.

8.6 IN-WAVEGUIDE MEASUREMENTS OF MMW DIELECTRIC PROPERTIES OF CERAMIC MATERIALS FOR THE U.S. FUSION REACTOR MATERIALS RESEARCH PROGRAM -- James C. Kennedy, III, Eugene F. Farnum, and Frank W. Chard, Jr. (Los Alamos National Laboratory) **302**

As part of an International Energy Agency (IEA) collaboration, a set of round-robin materials was purchased for comparing dielectric measurements at laboratories in the United Kingdom (UK). Spain. Germany, U.S., and Japan. P. Pells at Aldermasten, UK, purchased MACOR 9658, a glass-mica composite, and Roger Stoller from Oak Ridge National Laboratory, purchased WESGO AL-300 and AL-995, polycrystalline alumina standards. We obtained some of each of these materials for making these measurements. Our results have been shared with the other IEA partners, and P. Pells is preparing a summary document. We used the millimeter wave apparatus described below and elsewhere in detail to measure the dielectric properties of these materials at 90 to 100 GHz at room temperature. The nominal purity of AL-300 was 0.967; the nominal purity of AL-995 was 0.995. Our method was to measure the power transmission coefficient. We used computerized data reduction techniques to compute k (the dielectric constant) and tan-delta (the loss tangent) directly from transmission maxima and their corresponding frequencies; to verify this method, we applied the same technique to theoretically derived channel spectra that were obtained by solving exactly the complex transmission coefficient. given k and tan-delta. The alumina material with a lower level of purity resulted in higher loss but lower dielectric constant. We obtained dielectric constants that were higher for all the materials than manufacturer-reported values taken at lower frequencies. In addition, we obtained higher dielectric constant values than those found by other investigators at 100 GHz for AL-995 and MACOR. Tan-delta values were in good agreement with those of other investigators obtained by free-space methods and dispersive Fourier-transform techniques in the same frequency range.