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Capsules HFIR-MFE-JP9 through 16 were installed in the High Flux Isotope Reactor (HFIR) target July 20, 1990 for irradiation beginning with HFIR fuel cycle 289. Of these eight target capsules, JP10, 11, 13, 16 completed their scheduled number of cycles (11) and were removed from the reactor in September 1991. In addition, JP14 was removed from the reactor at the end of cycle 310 (September 18, 1992) after 21 cycles. Capsules JP9, 12, and 15 have completed 32 of 37 cycles.

Target capsules JP17, 18 and 19 contained fracture toughness specimens and were irradiated to approximately 3 dpa. All three capsules have been disassembled and testing is in progress.

Three new target capsules, JP20, 21, and 22, have teen assembled and are scheduled to begin irradiation at the start of Cycle 322. These capsules will operate at 300-600°C and will achieve doses of 8, 18 and 35 dpa, respectively. Many isotopically tailored specimens are included in the matrix.

The RB* experiments are the continuation of the ORR spectrally tailored experiment. In this stage, a hafnium liner is used to modify the spectrum to simulate fusion helium to displacements per atom (He/dpa) ratio in the austenitic steel specimens. Of the four capsules in this set. two (60J-1 and 330J-1) are ready to be disassembled, having achieved a total fluence of 18 dpa, and two (200J-1 and 400J-1) are in the reactor.

1.2 FABRICATION AND OPERATION OF HFIR-MFE RB* SPECTRALLY TAILORED IRRADIATION CAPSULES—A. W. Longest J. E. Pawel. D. W. Heatherly, R. G. Sitterson, and R. L. Wallace (Oak Ridge National Laboratory). 23

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 test specimens were 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 outer sleeve and containment tube.

Irradiation of the 60 and 330°C capsules, which started on July 17, 1990, was completed on November 14, 1992, after 24 cycles of irradiation to an incremental damage level of approximately 10.9 displacements per atom (dpa). Assembly of the follow-up 200 and 400°C capsules was completed in November, 1992, and their planned 20-cycle irradiation to approximately 9.1 incremental dpa was started on November 21, 1992. As of September 4, 1993, the 200 and 400°C capsules had successfully completed nine cycles of irradiation to approximately 4.1 incremental dpa.

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2.2 TRANSMUTATION OF Mo, Re, W, Hf, AND V IN VARIOUS IRRADIATION TEST FACILITIES AND STARFIRE--L. R. Greenwood and F. A. Garner (Pacific Northwest Laboratory) 60

With the exception of V and Cu alloys, the formation of solid transmutants has not been thought to play a large role in the response of metals to irradiation. For a number of refractory elements, however, it appears that very large levels of solid transmutation can occur both in fission and fusion spectra. The phase stability and other properties of alloys can be strong functions of the changing composition. The spectral sensitivity of transmutation also poses some difficulty in correlating data from various neutron spectra. Transmutation of Mo, Re, W, Hf and V have been calculated for the Fast Flux Test Facility (FFTF), the High Flux Isotope Reactor (HFIR), and the STARFIRE fusion reactor. Results show that Re and W undergo sizable transmutation in these spectra, and may limit the use of these elements in fusion reactor materials. Vanadium and hafnium have high transmutation rates in HFIR and may require spectral tailoring.

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5.1 MOLECULAR DYNAMICS CALCULATIONS OF DEFECT ENERGETICS IN β-SiC*--Hanchen Huang and Nasr Ghoniem (Mechanical, Aerospace and Nuclear Engineering Department, University of California, Los Angeles). 75

The Molecular Dynamics (MD) method is used to calculate defect energetics in β-silicon carbide. Many body interaction effects in this covalent material are accounted for by using a hybrid of two body and three-body potentials. Pearson's potential is modified to accurately fit sublimation energy of β-SiC. Inter-atomic potentials among silicon, carbon, and helium

atoms are also constructed based on available data and reasoning. A micro-crystal is constructed to represent the computational cell and external forces are applied to the micro-crystal so that it behaves as a part of an infinite medium. Potential energy for the unperturbed computational cell is first calculated. The cell is then set at a defect configuration and relaxed, and the potential energy of the relaxed cell is calculated. The difference between the potential energy of the unperturbed cell and that of the defect-containing cell is used to calculate formation energies of point defects. defect clusters and helium-filled vacancy clusters in SiC. Binding energies, and migration energies of the point defects are then deduced. Pre-exponential factors of point defect diffusion coefficients are calculated from potential profile oblained. Calculated activation energies and pre-exponential factors of thermal diffusion through vacancy mechanism are compared to experimental data. Good agreement backups our construction of potential functions and MD calculations.

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H. L. Heinisch (Pacific Northwest Laboratory). B. N. Singh (Risø National Laboratory) and
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5.3 INFLUENCE OF HELIUM GENERATION RATE AND TEMPERATURE HISTORY ON MECHANICAL PROPERTIES OF MODEL Fe-Cr-Ni ALLOYS IRRADIATED IN FFTF AT RELATIVELY LOW DISPLACEMENT RATES--M. L. Hamilton, F. A. Gamer and
D. J. Edwards (Pacific Northwest Laboratory). 94

In agreement with earlier studies conducted at higher displacement rates. the evolution of mechanical properties of model Fe-Cr-Ni alloys irradiated at lower displacement rates in the ^{59}Ni isotopic doping experiment does not appear to he strongly affected by large differences in helium generation rate. This insensitivity to helium/dpa ratio is exhibited during both isothermal and non-isothermal irradiation. The overall behavior of the model alloys used in this study is dominated by the tendency to converge to a saturation strength level that is independent of thermomechanical starting state and helium/dpa ratio, but which is dependent on irradiation temperature and alloy composition.

5.4 MICROSTRUCTURAL EVOLUTION OF NEUTRON-IRRADIATED Fe-Cr-Ni TERNARY ALLOYS AT HIGH HELIUM GENERATION RATES—N. Sekimura (University of Tokyo)
and F.A. Gamer (Pacific Northwest Laboratory). 102

A series of Fe-Cr-Ni ternary alloys were irradiated to 12-14 dpa in the Oak Ridge Research Reactor (ORR). In this experiment, most of helium atoms were produced by the ^{58}Ni(n,γ)^{59}Ni(n,α)^{56}Fe sequential reaction, and therefore, the helium generation rate was almosl proportional to the nickel content, ranging from 27 to 58 appm He/dpa in 20 to

45 wt.%Ni alloys. Irradiation temperatures were 330, 400, 500 and 600°C, and the specimen variables were the nickel level, the chromium level and cold working. The high levels of helium, combined with details of the temperature history, were found to have dominated to microsuuctural development, sometimes overwhelming and even reversing the action of a given material variable. Microscopy examination showed that there was an unprecedented refinement of cavity microstructure at all temperatures. which accounts for the substantial alterations in both swelling and mechanical properties observed in earlier studies.

5.5 MICROSTRUCTURAL DEVELOPMENT IN IRRADIATED VANADIUM ALLOYS—
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Three simple vanadium alloys, V-5Al, V-1Ni, and V-2Ti-1Ni have been examined to determine the effects of fast neutron irradiation in microsuuctural evolution. Specimens were irradiated in EBR II at temperatures in the range 425 to 600°C to doses of 15 and 31 dpa. Each alloy responded very differently to irradiation. All V-5AI specimens were generally void-free and contained high densities of coherent precipitates, and moderate densities of network dislocations. Swelling was much higher in V-1Ni. Voids were uniformly distributed, but with widely varying shape and size. Precipitation of two types also developed. The response in V-2Ti-1Ni was intermediate to that of the other two alloys. Irradiation produced extensive precipitation of thin rods and a moderate density of large voids often associated with much larger precipitate particles. The dislocation structure strongly interacted with the rod precipitate particles.

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6.1.1 REPORT ON IEA COLLABORATION ON FERRITIC/MARTENSITIC STEELS --
R. L. Klueh (Oak Ridge National Laboratory). 129

A collaborative program is underway under the auspices of the IEA (International Energy Agency) to develop a reduced-activation ferritic steel for a demonstration fusion power plant A task has been initialed for which Japan will produce a 5000-10000 kg heat of reduced-activation 8Cr-2WVTa steel. Once the steel is produced, the Japanese, EC, and US fusion programs will conduct detailed cooperative test programs on the steel. Two meetings have been held to plan the collaboration, and the results from those meetings are reviewed.

6.1.2 EFFECT OF LOADING MODE ON THE FRACTURE TOUGHNESS OF A FERRITIC/ MARTENSITIC STAINLESS STEEL—Huaxin Li (Associated Western Universities—Northwest Division), R. H. Jones (Pacific Northwest Laboratories), J. P. Hirth (Washington Stale University--Pullman) and D. S. Gelles (Pacific Northwest Laboratories). 134

The critical J integrals of mode 1 (J_{IC}), mixed-mode I/III (J_{MC}), and mode III (J_{mc}) were examined for a ferritic stainless steel (F-82H) at ambient temperature. A determination of J_{MC} was made using modified compact-tension specimens. Different ratios of tension/shear stress were achieved by varying the principal axis of the crack plane between 0 and 55 degrees from the load line. J_{mc} was determined using triple-pantleg like specimens. The results showed that F-82H steel is a very tough steel. Both J_{IC} and J_{mc} are about 500 kJ/m^2, and mode I tearing modula (TI) is about (360 kJ/m^2)/mm. However, J_{MC} and mixed-mode tearing modula (T_M) values varied with the crack angles and were lower than

their mode I and mode III counterparts. Both the minimum J_{MC} and T_M values occurred at a crack angle between 40 and 50 degrees, where the load ratio of $\sigma i/\sigma iii$ was 1.2 to 0.84. The J_{min} was 240 kJ/m^2, and ratios of J_{IC}/J_{min} and J_{mc}/J_{min} were 2.1 and 1.9, respectively. The morphology of fracture surfaces was consistent with the change of J_{MC} and T_M values. While the upper shelf-fracture toughness of F-82H depends on loading mode the J_{min} remains very high. Other important considerations include the effect of mixed-mode loading on the DBT temperature, and effects of hydrogen and irradiation on J_{min}.

6.1.3 IMPACT TOUGHNESS OF IRRADIATED REDUCED-ACTIVATION FERRITIC STEELS --R. L. Klueh and D. J. Alexander (Oak Ridge National Laboratory). . . . 148

Eight chromium-tungsten steels ranging from 2.25 to 12 wt% Cr were irradiated at 365°C to 13-14 dpa in the Fast Flux Test Facility. Post irradiation Charpy impact tests showed a loss of toughness for all steels, as measured by an increase in the ductile-brittle transition temperature (DBTT) and a decrease in the upper-shelf energy. The most irradiation-resistant steels were two 9% Cr steels: the DBTT of a 9Cr-2W-0.25V-0.1C steel increased 29°C, and for the same composition with an addition of 0.07% Ta the DBTT increased only 15°C. This is the smallest shift ever observed for such a steel irradiated to these levels. The other steels developed shifts in DBTT of 100 to 300°C. A 2.25% Cr steel with 2% W, 0.25% V, and 0.1% C was less severely affected by irradiation than 2.25% Cr steels with 0.25% V and no tungsten. 2% W and no vanadium, and with 1% W and 0.25% V. Irradiation resistance appears to be associated with microstructure, and microstructural manipulation may lead to improved properties.

6.1.4 MICROSTRUCTURAL DEVELOPMENT IN REDUCED ACTIVATION FERRITIC ALLOYS IRRADIATED TO 200 DPA AT 420°C--D. S. Gelles (Pacific Northwest Laboratory) 156

Density change and microstructural development are reported for nine reduced activation ferritic steels covering the range 2.3 to 12 Cr with varying additions of V and/or W for hardening and up to 6.5 Mn for austenite stability. Specimens were examined following irradiation in FFTF/MOTA at 420°C to a dose exceeding 200 dpa. Void swelling was found, but the swelling remained at 5% or below, with the worst case in an alloy of 9Cr-2Mn-1WV. The carbide structure pinning Martensite lath boundaries remains in place.

6.1.5 IRRADIATION CREEP AND SWELLING OF THE U.S. FUSION HEATS OF HT9 AND 9Cr-1Mo TO 208 DPA AT –400°C–M. B. Toloczko (University of California at Santa Barbara), F. A. Gamer (Pacific Northwest Laboratory)[a], and C. R. Eiholzer (Westinghouse Hanford Company). 164

The irradiation creep and swelling behaviors of the fusion heats of HT9 and 9Cr-1Mo at ~400°C have been measured to exposures as large as 208 dpa, using both diametral and density measurements of helium-pressurized creep tubes. Void swelling was found in both alloys at 208 dpa to occur at rates of 0.012%/dpa or less, with the swelling of HT9 exhibiting a larger degree of stress enhancement than 9Cr-1Mo. The creep rate of HT9 is rather nonlinear in its response to keep stress level in the range 0-200 MPa, but 9Cr-1Mo exhibits only slightly greater than linear behavior with stress level. The creep-swelling coupling coefficients for 9Cr-1Mo are consistent with values obtained for other steels.

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Although the design of the International Thermonuclear Experimental Reactor is still in an evolutionary phase, the most probable choice for a structural material is a 300 series austenitic stainless steel. An experiment was carried out in the High Flux Isotope Reactor to quantify the effects of neutron irradiation on the fracture toughness properties of a range of austenitic stainless steels. The alloys investigated were provided by programs in the European Community. United States, and Japan; they included 316, 316L, and titanium-stabilized steels in cold-worked, annealed, and welded conditions. Subsize disk compact tension specimens were irradiated to a dose of about 3 dpa and 50 appm helium (the expected fusion value) at temperatures of either 60 to 125°C or 250 to 300°C. With the exception of a cold-worked air-melted heat of 316, all materials retained excellent fracture toughness, with K_c values varying from 150 to 400 MPa$\sqrt{m}$ over the test temperature range from 90 to 250°C.

6.2.2 TENSILE PROPERTIES OF BASE AND WELD METAL SPECIMENS OF AUSTENITIC STAINLESS STEELS IRRADIATED IN THE HIGH FLUX ISOTOPE REACTOR—K. Shiba and S. Jitsukawa (Japan Atomic Energy Research Institute) and J. E. Pawel and A. F. Rowcliffe (Oak Ridge National Laboratory). ..... .181

Tensile properties of Japanese Primary Candidate Alloy (JPCA) base metal and welded specimens imdiated in HFIR target positions were investigated. Electron beam (EB) and tungsten inert gas (TIG) welding techniques were used to make welded specimens. Welded specimens included weld metal and weld joint specimens. These specimens were irradiated up to 18 dpa at 300, 400, 500 and 600°C. JPCA solution annealed base metal had maximum irradiation hardening at 300°C, and it showed minimum elongation at that temperature. EB welded specimens exhibited almost the same tensile properties as the base metal, with the strength of the EB weld metal and weld joint specimens being only slightly lower than that of the base metal. EB welded specimens retained uniform elongations in the range of 2 to 7% after irradiation. TIG welded specimens had less ductility than the EB welded specimens, however the other tensile properties of the TIG welded specimens were almost same as for the EB welded specimens.

6.2.3 A PRELIMINARY ANALYSIS OF CRACK GROWTH IN TYPE 316L STAINLESS STEEL IN AIR AND WATER UNDER CYCLIC LOADING CONDITIONS FOR ITER FIRST-WALL DESIGN—M. R. Fox, W. K. Soppet, W. E. Ruther, and T. F. Kassner (Argonne National Laboratory). 187

Initial results on stress corrosion cracking (SCC) susceptibility and crack growth rate (CGR) of Type 316L SS in oxygenated water were used with fracture-mechanics relations to compare the fatigue life (cycles to failure) in air and water at 200°C of a coolant tube within the ITER first-wall structure for burn and pulse times of 480 and 1000 s, respectively. Fatigue life was evaluated as a function of maximum stress intensity. initial flaw depth in the 3.2-mm wall of the tube, water purity, and threshold stress intensity for SCC.

6.2.4 DENSITY MEASUREMENTS PERFORMED ON ISPRA SECOND GENERATION AMCR ALLOYS IRRADIATED IN MOTA-2A AND MOTA-2B--F. A. Gamer (Pacific Northwest Laboratory), P. Schiller (Ispra Establishment), and H. Takahashi (Hokkaido University). 198

Density change measuremenls have been completed on the Ispra second generation AMCR alloys irradiated in MOTA-2A and MOTA-2B. The various compositional modifications induce a wide variation of swelling behavior in the range 423-600°C at 32-61 dpa.

6.2.5 DENSITY CHANGES INDUCED BY NEUTRON IRRADIATION IN DYNAMICALLY COMPACTED TUNGSTEN AND PCA IN MOTA's 2A and 2B--F. A. Gamer (Pacific Northwest Laboratory) and J. Megusar (Massachusetts Institute of Technology). 201

Dynamically compacted tungsten with a starting density of 95.3% of the theoretical value densified 2 to 3% when irradiated in FFTF/MOTAs 2A and 2B at three temperatures between 423 and 600°C and displacement levels corresponding to 32 to 36 dpa in stainless steel during the first irradiation segment and 54.2 to 60.8 dpa during the second segment. Rapidly solidified and dynamically compacted PCA with high levels of titanium and carbon were also irradiated at these conditions. The density changes were variable between nominally identical specimens but were small enough to determine that significant swelling had not occurred. Microscopy is necessary to determine whether any void growth occurred in addition to precipitate-related strains.

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6.3.1 EFFECTS OF FABRICATION VARIABLES ON IMPACT PROPERTIES AND MICROSTRUCTURE OF V-CR-TI ALLOYS—H. M. Chung. J. Gazda, L. J. Nowicki, J. E. Sanecki. and D. L. Smith (Argonne National Laboratory). **207**

In spite of the small differences in composition of alloying elements, V-4Cr-4Ti, V-5Cr-5Ti, and V-5Cr--5Ti alloys exhibited surprisingly large variations in ductile-brittle transition temperatures (DBTTs) (<-190, -60, and -85°C. respectively) in previous investigations. To provide a better understanding of the origin of the variation, effects of fabrication parameters (extrusion, forging. annealing temperature, and quality of Ti raw material) on the microstructure and impact properties were investigated in this work. Fracture surfaces of V-5CR-5Ti and V-5Cr-3Ti alloys, which exhibited DBTTs –110°C higher than that of V-4Cr-4Ti, were characterized by relatively large precipitates (2,000-17,000 nm) which appear to be complex vanadium compounds containing different compositions of C, S, O, Cl, Ca, Na, K, Cr, and Cu. The two alloys also contained high-density precipitates 30-70 nm in size that seem to be vanadium compounds rich in O, C, S, and Ca. This unusual precipitation of vanadium compounds in the V-5Cr-5Ti and V-5CR-3Ti alloys seems to be associated in part with Ti sponge (relatively rich in Cl, Na, and K) that was used to prepare the alloy ingots. Based on thermodynamic consideration, formation of some of the complex precipitates (e.g., vanadium oxycarbochloride) is preferred to titanium oxycarbonitirides in alloys containing a high level of Cl. Vanadium-based precipilates were absent in V-4Cr-4Ti, which was fabricated using vacuum-arc-melted Ti plates low in Cl, Na, and K. V-5Cr-3Ti, which was forged at ≈650°C rather than extruded at the usual temperature of 1150°C. also contained unusually small titanium oxycarbonitrides in high-density. Cleavage was the predominant process in brittle fracture of V-5Cr-5Ti. However, evidence of intergranular separation, apparently associated with grain-boundary segregation of Ti, S, and P, was also observed in the material. The origin of the higher than-expected DBTTs of the V-5Cr-5Ti and V-5Cr-3Ti alloys seems to be associated with the unusual precipitation of vanadium compounds and intergranular separation. Effect of post-fabrication annealing on ductile-brittle-transition behavior was secondary. although annealing at ≈1050°C produced the most favorable impact properties. In contrast to V-5Cr-5Ti and V-5Cr-3Ti, V-4Cr-4Ti remained ductile for impact temperatures higher than -196°C regardless of annealing at 1050 or 1125°C.

6.3.2 EFFECT OF NEUTRON IRRADIATION ON TENSILE PROPERTIES OF V-Cr-Ti Alloys--B. A. Loomis, L. J. Nowicki, and D. L. Smith (Argonne National Laboratory). 219

Vanadium-base alloys with Cr and Ti as the principal alloying additions have a number of attractive physical and mechanical properties for application as suuctural material in the International Thermonuclear Experimental Reactor (ITER) and in fusion reactors that may be constructed in the future. These alloys offer an excellent combination of low, long-term radioactivation, high-temperature capability, and resistance to irradiation damage. At the present time, the V-4Cr-4Ti alloy is recommended for the near-optimal combination of desirable physical and mechanical properties for a vanadium alloy structural material. In this paper, we present the tensile properties of the V-4C4-4Ti alloy and, in addition. tensile properties of V-14Cr-5Ti, V-9Cr-5Ti, V-3Ti-1Si, V-5Ti, and V-18Ti alloys after neutron irradiation to 28–46 atom displacements per atom (dpa) at 420,520, and 600°C.

6.3.3 IRRADIATION-INDUCEDDENSITY CHANGE AND MICROSTRUCTURAL EVOLUTION OF VANADIUM-BASE ALLOYS—H. M. Chung, B. A. Loomis, L. Nowicki, J. Gazda, and D. L. Smith (Argonne National Laboratory). 223

Irradiation-induced density change (swelling) and microstructural evolution of V-Ti, V-Cr-Ti. and V-Ti-Si alloys were investigated after irradiation in a fast reactor at 420 to 600°C up to 114 dpa. The alloys exhibited swelling maxima in the damage range of 30 to 70 dpa and swelling decreased on irradiation to higher dpa. The swelling resistance of the alloys was associated with high-density precipitation of ultrafine Ti_5Si_3, and it was concluded that Ti>4 wt.% and 400 to 1000 wppm Si are desirable to effectively suppress swelling. Swelling resistance of V-4Cr-4Ti was excellent. and this alloy is virtually immune to irradiation embrittlement. The excellent swelling resistance was associated with ultrafine Ti_5Si_3 for irradiation at 520 to 600°C and dense formation of dislocation loops for irradiation at <420°C.

6.3.4 RELATIONSHIP OF MICROSTRUCTURE AND MECHANICAL PROPERTIES--J. Gazda, B A. Loomis, and L. J. Nowicki (Argonne National Laboratory), and S. Danyluk (University of Illinois at Chicago), ... ,232

The relationship of composition, number density, and diameter of precipitates in the microstructures of unalloyed V and V-Cr-Ti alloys to the yield strength, hardness. creep stress, and ductile-brittle transition temperature (DBTT) of these materials was determined from electron microscopy analyses of precipitates in these materials and from mechanical property data. Unalloyed V and V-Cr-Ti alloys with ≤3 wt.% Ti contained VC and Ti(CNO) precipitates, which were coherent with the matrix. The most common precipitates in the alloys were $Ti(C_{1-x-y}N_xO_y)$ that were noncoherent with the matrix. The number density of noncoherent precipitates was maximal in V-3Ti and and V-5Cr-3Ti alloys, and the average diameter of noncoherent precipitates was minimal in V-(1-3)Ti and V-5Cr-3Ti alloys. The increase of yield strength and hardness of V on alloying with Ti and Cr was shown to be primarily due lo coherent precipitates, solute-atom misfit, and shear-modulus difference effecu. The creep stress for rupture in 1000 h was related lo the number of precipitates. whereas the DBTT was related to the volume fraction of precipitates.

6.3.5 MICROSTRUCTURAL STABILITY OF V-4Cr-4Ti DURING NEUTRON IRRADIATION--H. Matsui and M. Tanno (Tohoku University) and J. Gazda and H. M. Chung (Argonne National Laboratory). 240

Microstructural evolution of V-4Cr-4Ti, an alloy recently shown to exhibit excellent tensile and creep properties. virtual immunity to irradiation embrittlement, and good resistance to swelling, was characterized after irradiation in a lithium environment in the Fast Flux Text Facility (FFTF) (a sodium-cooled fast reactor located in Richland, Washington) at 420, 520, and 600°C to 24-34 dpa. The primary feature of microstructural evolution during irradiation

at 520 and 600°C was high-density formation of ultrafine Ti_5Si_3 precipitates and short dislocations. For irradiation at 420°C, precipitation of Ti_5Si_3 was negligible, and "black-dot" defects and dislocations were observed in significantly higher densities. In spite of their extremely high densities, neither the "black-dot" defects nor Ti_5Si_3 precipitates are overly detrimental to ductility and toughness of the alloy, yet they very effectively suppress irradiation-induced swelling. Therefore, these features. normally observed in V-base alloys containing Ti and Si, are considered stable. Unstable microstructural modifications that are likely to degrade mechanical properties significantly were not observed, e.g., irradiation-induced formation of fine oxides, carbides, nitrides, or Cr-rich clusters.

6.3.6 STATUS OF THE DYNAMIC HELIUM CHARGING EXPERIMENT (DHCE)--H. Tsai. H. M. Chung. B. A. Loomis, and D. L. Smith (Argonne National Laboratory). . . . . .,247

Irradiation of the seven DHCE capsules was completed in MOTA-2B at the end of FFTF Cycle 12B. The accrued exposure was 203.3 effective full power days (EFPDs) and the peak damage in the samples was ≈23 dpa. All seven capsules have been discharged from the FFTF and shipped to ANL-East. Procedures have been developed to disassemble the tritium-containing capsules to retrieve the specimens for examination and testing.

6.3.7 EFFECTS OF NEUTRON IRRADIATION AND HYDROGEN ON DUCTILE-BRITI'LE TRANSITION TEMPERATURES OF V-Cr-Ti ALLOYS—B. A. Loomis, H. M. Chung, L. J. Nowicki, and D. L. Smith (Argonne National Laboratory). 253

The effects of neutron irradiation and hydrogen on the ductile-brittle transition temperatures (DBTTs) of unalloyed vanadium and V-Cr-Ti alloys were determined from Charpy-impact tests on 1/3 ASTM-standard-size specimens and from impact tests on 3-mm diameter disks. The tests were conducted on specimens containing <30 appm hydrogen and 600-1200 appm hydrogen and on specimens after neutron irradiation to 28-46 atom displacements per atom at 420, 520. and 600°C. The DBTTs were minimum (<-220°C) for V-(1-5)Ti alloys and for V-4Cr-4Ti alloy with <30 appm hydrogen. The effect of 600-1200 appm hydrogen in the specimens was to raise the DBTTs by 60-100°C. The DBTTs were minimum (<-200°C) for V-(3-5)Ti and V-4Cr-4Ti alloys after neutron irradiation.

6.3.8 WELDING OF VANADIUM ALLOYS—D. R. Diercks, C. F. Konicek, and R. H. Lee (Argonne National Laboratory). 258

An exploratory activity bas been initiated to investigate potential welding processes and the effects of these processes on the properties of candidate vanadium-base alloys has been initiated. Weldments of V-5Cr-5Ti and V-15Ti-7.5Cr have been prepared by the tungstenlinert-gas process without the use of filler metal. The weldments were examined metallographically and found to be free of significant defects. Chemical analyses of the weldments indicate only slight oxygen pickup during welding, and microhardness profiles indicate no unusual effects.

6.3.9 DEVELOPMENT OF MINIATURE-SPECIMEN TECHNOLOGY TO DETERMINE DUCTILE-BRITI'LE TRANSITION BEHAVIOR OF IRRADIATED VANADIUM ALLOYS—H. M. Chung. L. J. Nowicki, J. Gazda, D. E. Busch, and D. L. Smith (Argonne National Laboratory). 263

Simple uninstrumented impact tests were conducted on unirradiated and irradiated transmission electron microscopy (TEM) disks of vanadium alloys between -196 and 300°C. The results were combined with quantitative fractography of the fractures specimens in a scanning electron microscope (SEM) to determine the effect of impact temperature on fracture behavior. Based on the fractographic information. ductile-brittle transition behavior of several alloys was determined before and after irradiation. The ductile-brittle transition

temperature (DB'IT) obtained from this technique was then compared with the DB1T from Charpy impact testing of the same material. A correlation curve was then obtained for use in estimating DBTTs of irradiated vanadium-base alloys based on ductile-brittle transition behavior of standard-size TEM disks. The method seems to be adequate to estimate the ductile-brittle transition behavior of a material whose Charpy-impact DB1T is higher than approximately **-70°C.** For a material whose Charpy DB'IT is <-70°C, the method can provide an estimation of upper limit of the DB'IT. Compared to **a** similar; method based on slow-strain-rate small-punch testing, the present method seems to be significantly more advantageous in evaluating ductile-brittle transition behavior.

6.3.10 ASSESSMENT OF NIOBIUM-BASE ALLOYS FOR STRUCTURAL APPLICATIONS IN THE ITER DIVERTOR—I. M. Purdy (Argonne National Laboratory). **268**

The corrosion and embrittlement of pure Nb, Nb-1Zr, Nb-5Mo-1Zr, and Nb-5V-1.25Zr (alloy elements in wt.%) were evaluated in high-purity (HP) deoxygenated water at ***300°C*** for **up** to 120 days. One heat of the Nb-5V-1.25Zr alloy ("O" lot) exhibited both a modest corrosion rate and good resistance to emhrittlement relative to other Nb-base alloys. At present, Nb-5V-1.25Zr is the most promising Nb-base alloy with respect to both corrosion and embrittlement characteristics in HP deoxygenaled water at ***300°C.***

6.3.11 THE INFLUENCE OF STARTING **STATE** ON NEUTRON INDUCED DENSITY CHANGES OBSERVED IN Nb-Zr AND Mo-41Re AT HIGH EXPOSURES--
F. A. Gamer, L. R. Greenwood, and D. J. Edwards (Pacific Northwest Laboratory). 274

Both Nb-1Zr and Mo-41Re were irradiated in FFTF at five temperatures between 420 and 730°C, reaching maximum exposure levels of 58 lo 111 dpa, depending on the irradiation temperature. Each alloy was irradiated in two starting conditions. cold-worked and aged or annealed and aged. The neutron-induced swelling behavior in response to starring condition was quite different for the two alloys and appeared to reflect a strong role of precipitation in each. While transmutation does not play a major role in the response of Nb-1Zr, it plays a significant role in the behavior of **Mo-41Re,** which becomes a heavily-precipitated Mo-Re-Os-Tc-Ru alloy. This strong response *may* preclude the use of Mo-Re alloys for fusion application.

6.4 COPPER ALLOYS. 283

6.4.1 NEUTRON-INDUCED MICROSTRUCTURAL ALTERATION OF GLIDCOP™ ALLOYS AT 415°C AND HIGH NEUTRON EXPOSURE--D. J. Edwards. F. A. Gamer (Pacific Northwest **Laboratory)**[a], J. W. Newkirk (University of Missouri) and **A.** Nadharni (SCM Metal Products). 285

GlidCop™ internally oxidized copper alloys remain the leading candidates for high heat flux applications in fusion reactors. This paper presents the microstructural changes incurred in three GlidCop™ alloys exposed to long term. high temperature neutron irradiation.

Irradiation at high lemperature produced **a** microstructure containing a much lower dislocation density than the unirradiated specimens. Although 1-50 nm size triangular oxide panicles were observed in areas with a very low number density of particles, spherical oxide panicles on the order of 5-7 nm in diameter, thought to be $CuAl_2O_4$, were the predominant morphology. The changes in grain size distribution. dislocation density, and precipitate type and distribution saturate in the range of 34 to **50** dpa. **as** reflected in the saturation of mechanical properties.

6.4.2 DISTRIBUTION OF TRANSMUTANT NICKEL FORMED IN FAST NEUTRON IRRADIATED COPPER—T. Muroga (Kyushu University) and F. A. Gamer (Pacific Northwest Laboratory)[a]. 293

Nickel formed by transmutation of pure copper after irradiation to 95.4 dpa in FFTF was measured using the EDS technique. The nickel measurements agree rather well with the predicted level, although the foil-averaged value is higher than the bulk-averaged value due to the large amount of void intersections that occur in this specimen, which swelled ~45%. Nickel was shown to segregate strongly at void surfaces.

6.5 ENVIRONMENTAL EFFECTS IN STRUCTURAL MATERIALS. 299

6.5.1 CONSIDERATION OF TRITIUM AND HYDROGEN DISTRIBUTIONS IN LAYERED STRUCTURES FOR ITER--E. P. Simonen, J. L. Brimhall and R. H. Jones (Pacific Northwest Laboratory). 301

The distributions of tritium from the plasma and hydrogen from the coolant in candidate ITER structures depends on the selection of base alloys as well as layered materials that are applied to improve compatibility with the plasma or the coolant. Two hydrogen sources are considered in the present calculations using the DIFFUSE 83 code, namely, implantation-driven permeation of tritium from the plasma and gas-driven permeation of hydrogen from the coolant. A stainless steel layer in contact with the coolant can achieve significant tritium and hydrogen concentrations depending on choice of base material, temperature, ion implantation flux and coolant pressure of hydrogen. Hydrogen ingress through a stainless steel layer 10 V structures may be significant but ingress through layers 10 Cu-based alloys is not expected to be significant.

6.5.2 MEASUREMENT OF ELECTRICAL RESISTIVITY OF THERMALLY GROWN TITANIUM NITRIDE THIN FILMS IN LIQUID LITHIUM--J.-H. Park and T. Domenico (Argonne National Laboratory). 312

A sample of pure Ti and a pair of Ti electrodes were placed in small capsules containing liquid Li and Li_3N for seven days at 710°C to investigate the formation of titanium nitride (TiN). An attempt was made to enhance the resistivity of in-situ-formed TiN films by adding to the Li small amounts of Al, Si, and Mg. which might be incorporated into the

films. One sample and set of electrodes was nitrided in Li containing Li_3N and another was immersed in pure Li. The electrical resistance of the films was –1.0 to 1.5 Ω and the values increased slightly with temperature, which is indicative of metallic conduction. Energy-dispersive-spectroscopy (EDS) analyses of the films by scanning electron microscopy (SEM) showed that none of the dopants was incorporated into the TiN lattice.

6.5.3 DEVELOPMENT OF ELECTRICAL-INSULATOR COATINGS: IN-SITU ELECTRICAL RESISTANCE MEASUREMENTS ON V-5%Cr-5%Ti IN LIQUID LITHIUM--J.-H. Park and G. Dragel (Argonne National Laboratory). 321

The electrical resistance of insulator coatings produced on V-5%Cr-5%Ti by exposure of the alloy 10 liquid Li that contained 5 at.% N, with and without 5 at.% dissolved Al, was measured as a function of time at temperatures between 250 and 500°C. The solute elements (N and Al) reacted in liquid Li with the alloy substrate at 415°C to produce thin adherent coatings. The resistance of the coating layer was –1.5 and 1.0 Ω at 415 and 500°C. respectively. Thermal cycling between 250 and 415°C did not change the resistance of the coating layers. These results and those reported previously suggest that homogeneous coatings can be produced on variously shaped surfaces by controlling the exposure time,

temperature. and composition of the liquid metal. The integrity of the coatings does not appear to be sensitive to defects (e.g., open pores, fissures, or microcracks) present in the alloy substrate in liquid Li.

6.5.4 INVESTIGATION OF A MAGNETIC TRAP USED IN A LITHIUM/STAINLESS STEEL TEST LOOP-- P. R. Luebbers and O. K. Chopra (Argonne National Laboratory). 326

The magnetic trap from a plugged cold-trap purification loop of a forced-circulation lithium system was examined to study mass transfer/deposition behavior in liquid lithium systems. Two types of particles, large faceted crystals and globules, were observed in the residue collected from the magnetic trap. These particles are similar to the (Mn,Fe) Ni and (Mn,FE) Ni3 compounds that had been identified earlier in the particulate collected from plugged sections of a cold-trap loop. The ternary nitride Li_9CrN_5 was also identified in the residue. Results indicate that a magnetic trap is effective for capturing magnetic particles within a liquid lithium loop.

6.5.5 COMPATIBILITY OF OXIDIZED TYPE **316** SS WITH STATIC GALLIUM—P R. Luebbers, W. F. Michaud, and O. K. Chopra (Argonne National Laboratory). **333**

Scoping tests were conducted on compatibility of gallium with Type **316** SS in three conditions: as-received. oxidized in water at 290°C for 100 h. and oxidized in air at 350°C for 100h. Corrosion tests were conducted at 400°C for limes up to 3000 h. The results indicate that at **400°C,** an iron[II] chromite coating does not provide protection against corrosion. The oxide coating dissociates to form β-Ga_2O_3, and corrosion of the alloy proceeds by dissolution accompanied by formation of $FeGa_3$ and other gallium/intermetallic compounds. However, oxidized specimens have a lower corrosion rate than nonoxidized Type **316** SS. Corrosion rates are **4.0. 2.2,** and **2.7** mm/y for Type **316** SS as-received. oxidized in water, and oxidized in air, respectively.

7.0 SOLID BREEDING MATERIALS AND BERYLLIUM. 341

7.1 RADIATION EFFECTS IN BERYLLIUM USED FOR PLASMA PROTECTION-- D. S. Gelles (Pacific Northwest Laboratory). M. Dalle Donne (Kernforschungszentrum Karlsruhe, Germany). G. A. Sernyaev (SF NIKIET, Russia Federation) and H. Kawamura (Blanket Irradiation & Analysis Laboratory, JAERI, Japan). . . . . ,**343**

This paper reviews the literature on beryllium, emphasizing the effects of irradiation on essential properties. Swelling and embrittlement experiments as a function of irradiation temperature and dose, and as a function of neutron spectrum are described, and the results are quantified, where possible. Effects of impurity content are also reported, from which optimum composition specifications can be defined. Microstructural information has also been obtained to elucidate the processes controlling the property changes.

The available information indicates that beryllium divertors can be expected to embrittle quickly and many need frequent replacement.

7.2 POSTIRRADIATION EXAMINATION OF BEATRIX-II, PHASE I--O. D. Slagle. F. D. Mobbs. D. L. Baldwin (Pacific Northwest Laboratory), T. Takahashi, K. Ncda (Japan Atomic Energy Research Institute) R. A. Verrall (AECL Research). 358

BEATRIX-II is an in situ tritium recovery experiment that was designed to characterize the behavior of lithium ceramics irradiated to high burnup in a fast neutron flux. Postirradiation

examination was carried out on the Phase I venred canisters: one containing a Li_2O ring capable of temperature changes and the other a Li_2O solid specimen with a center temperature of 1000°C.

The tritium inventory of the ring specimen at 650°C was determined to be in the range from 0.2-0.6 wppm while for the solid specimen the inventory varied from 1.4 wppm at the surface to 0.06 wppm at the inner surface. Downstream transport of the Li_2O by the sweep gas was determined to be insignificant from analyses of acid rinses of selected canister surfaces. Densification and restructuring of the solid specimen during irradiation resulted in the development of a central annulus. Ceramography was used to characterize the columnar grain structure and the mechanisms involved in its evolution.

7.3 TRITIUM RELEASE FROM LITHIUM TITANATE, A LOW ACTIVATION TRITIUM BREEDING MATERIAL -- J. P. Kopasz and C. E. Johnson (Argonne National Laboratory) and J. M. Miller (AECL Research, Canada). 366

The goals for fusion power are to produce energy in as safe, economical, and environmentally benign a manner as possible. To ensure environmentally sound operation low activation materials should be used where feasible. The ARIES Tokamak Reactor Study has based reactor designs on the concept of using low activation materials throughout the fusion reactor. For the tritium breeding blanket, the choices for low activation tritium breeding materials are limited. Lithium titanate is an alternative low activation ceramic malerial for use in the tritium breeding blanket. To date, very little work has been done on characterizing the tritium release for lithium titanate. We have thus performed laboratory studies of tritium release from irradiated lithium titanate. The results indicate that tritium is easily removed from lithium titanate at temperatures as low as 600 K. The method of titanate preparation was found to affect the tritium release, and the addition of 0.1% H_2 to the helium purge gas did not improve tritium recovery.

7.4 TRITIUM RELEASE FROM BERYLLIUM DISCS AND LITHIUM CERAMICS IRRADIATED IN THE SIBELIUS EXPERIMENT -- C. E. Johnson and J. P. Kopasz (Argonne National Laboratory) and D. L. Baldwin (Pacific Northwest Laboratory). . . . . ,373

The SIBELIUS experiment was designed to obtain information on the compatibility between beryllium and ceramics, as well as beryllium and steel, in a neutron environment. Four capsules were used to examine beryllium/ceramic (Li_2O, $LiAlO_2$, Li_4SiO_4, and Li_2ZrO_3) and beryllium/steel (Types 316L and 1.4914) compacts. Isothermal anneal experiments have been run on representative beryllium and ceramic disks from each of the four capsules at 550°C to 850°C in steps of 100°C. The results indicate that tritium release from the beryllium did not exhibit burst release behavior, as previously reported, but rather a progressive release with increasing temperature. Generally –99 percent of the tritium was released by 850°C. Tritium release from the ceramic discs was quite similar to the behavior shown in other dynamic tritium release experiments on lithium ceramics. The tritium content in beryllium discs adjacent to a steel sample was found to be significantly lower than that found in a beryllium disc adjacent to a ceramic sample. Recoil of tritium from the ceramic into the beryllium appears to be the source of tritium entering the beryllium, probably residing in the beryllium oxide layer.

8.0 CERAMICS 379

8.1 TOUGHENING OF CERAMIC COMPOSITES WITH SLOW CRACK GROWTH–
C. R. Jones, C. H. Heneger, Jr., and R. H. Jones (Pacific Northwest Laboratory). 381

Ceramic matrix composites show the uncommon and promising property of increasing their toughness and maintaining their load carrying capacity with increasing crack lengths under conditions of subcritical crack growth (SCG). As the crack grows. a more extensive bridging zone is formed allowing it to withstand far greater stress intensities than would be expected from fracture tests. It is also possible that crack bifurcation contributes to this effect, but is not likely to be the primary cause. It has not been determined whether or not an equilibrium crack-wake bridging zone will occur in these samples. Further testing may show that it occurs at greater crack lengths than hose obtained in these tests or that it can't occur at all in these samples. Either way, the results of these tests are encouraging and expand the possible uses of **these** types of materials in applications that require long-term stability under stress.

8.2 POST-IRRADIATION ANALYSIS OF FIBER DEBONDING AND PULL-OUT IN SiC-SiC COMPOSITES--A.E-Azab and N. M. Ghoniem (University of California, Los Angeles). 389

The toughness of SiC-SiC composites is conuihuted by crack bridging, matrix crack deflection, fiber debonding and pull-out and other minor effects. Fiber pull-out toughening relies on the debonding and frictional characteristics of the fiber-matrix interface. The interface friction depends on the interface pressure (i.e., on misfit strains) and interface roughness. An earlier study by the present authors [1] has shown hat *misfit strain* in SiC-SiC composites undergo complex changes under neutron irradiation due to irradiation-induced creep and swelling. The present work complements the study presented in Ref. [1] by developing a calculational model for post-irradiation fiber debonding and pull-out toughness in SiC-SiC composites. It is shown that fiber debonding and pull-out toughness in SiC-SiC composites vary significantly with neutron fluence and irradiation temperature, which is a direct consequence of the dependence of the misfit strain on these irradiation variables.

8.3 RADIATION EFFECTS ON NICALON POLYMER PRECURSOR FIBERS --
L. L. Snead. M. C. Osborne and K. L. More (Oak Ridge National Laboratory). 396

This paper presents a summary of the radiation effects in the Nicalon polymer precursor fiber. It is shown that the serious degradation previously seen in this fiber **(and** fiber composite) is primarily due to the presence of a silicon oxycarbide phase. Results supporting this interpretation include microstructural analysis as well as post irradiation mechanical property measurements. Preliminary results on the effects ***of*** radiation on low oxygen Nicalon fibers are presented. The reduced oxygen fibers have shown radiation induced density, strength, Weibull's and Young's moduli changes typical for ceramic materials. This contrasts sharply with the poor irradiation behavior ***of*** the standard Nicalon fiber and suggests that improved radiation resistance ***can*** be expected in SiC/SiC composites fabricated with low oxygen Nicalon.

8.4 FATIGUE CRACK GROWTH RATE (FCGR) BEHAVIOR OF SiC/SiC COMPOSITES -- N. Miriyala, P. K. Liaw and C. J. McHargue (University of Tennessee, and L. L. Snead (Oak Ridge National Laboratory). 413

Two different techniques. viz. compliance and FRACTOMAT methods, have been evaluated to precisely determine the crack length during fatigue crack propagation testing. A software program has been developed to compute da/dN and AK values. Facilities are being augmented to conduct the tests at temperatures as high **as** 2000°C. Machining of SiC/SiC

composites has been successfully accomplished using the facilities at the University of Tennessee. Work has been initiated on the theoretical modeling of the various crack shielding mechanisms that can profoundly influence the crack propagation rates.

8.5 SUMMARY OF IEA WORKSHOP ON RADIATION EFFECTS IN CERAMIC INSULATORS--S. J. Zinkle (Oak Ridge National Laboratory, G. P. Pells (Harwell National Laboratory. England) and F. W. Chard, Jr. (Las Alamos National Laboratory). 417

A workshop on radiation effects in ceramics was held during the 6th International Conference on Fusion Reactor Materials in Stresa, Italy. Recent work on radiation effects in ceramics was reviewed by participants from Europe, Japan, and the United States. Most of the ensuing discussion centered on the phenomenon of radiation induced electrical degradation (RIED). A round robin RIED experiment on the IEA reference ceramic (Wesgo AL995) was formulated.

8.6 RADIATION-INDUCED ELECTRICAL DEGRADATION OF CERAMIC MATERIALS: AN ARTIFACT? -- W. Kesternich and F. Scheuermann (Institut fur Festkörperforschung, Germany), and S. J. Zinkle (Oak Ridge National Laboratory). 422

Recent electrical conductivity measurements made on aluminum oxide during irradiation with 28 MeV He+ ions near 550°C indicate that radiation induced electrical degradation (RIED) does not occur for damage levels up to –0.2 dpa. Increases in electrical current that resemble the RIED effect have been observed in some cases, but were found to be due to an increase in the surface leakage current and are attributed to a buildup of surface contamination during irradiation.

8.7 SEARCH FOR RADIATION-INDUCED ELECTRICAL DEGRADATION IN ALUMINA DURING SPALLATION-NEUTRON IRRADIATION -- E. H. Famum. F. W. Chard, Jr. W. F. Sommer, J. C. Kennedy III (Los Alamos National Laboratory) and Tatsuo Shikama (Tohoku University). 427

No summary submitted.

8.8 TRANSIENT EFFECTS OF IONIZING AND DISPLACIVE RADIATION ON THE DIELECTRIC PROPERTIES OF CERAMICS -- R. H. Goulding, S. J. Zinkle. R. E. Stoller, and D. A. Rasmussen (Oak Ridge National Laboratory). 434

A resonant cavity technique was used to measure the dielectric constant and loss tangent of ceramic insulators at a frequency near 100 MHz during gamma ray and pulsed fission reactor irradiation near room temperature. Tests were performed on single crystal and several different grades of polycrystalline Al_2O_3, $MgAl_2O_4$, AIN and Si_3N_4. Lead shielding experiments were performed for some of the pulsed fission reactor irradiations in order to examine the importance of gamma ray versus neutron irradiation effects. With the exception of AIN, the dielectric constant of all of the ceramics decreased slightly during the pulsed fission reactor irradiation. Significant transient increases in the loss tangent to values as high as 6×10^{-3} occurred during pulsed reactor irradiation with peak ionizing and displacements per atom (dpa) radiation fields of 4.2×10^4 Gh/s and 2.4 x 0-6 dpa/s, respectively. The loss tangent measured during irradiation for the different ceramics did not show any correlation with the preirradiation or postirradiation values. Analysis of the results indicates lhat the transient increases in loss tangent are due to radiation induced increases in the electrical conductivity. The loss tangent increases were proportional to the ionizing dose rate.

8.9 IN-CORE MEASUREMENT OF DC ELECTRICAL CONDUCTIVITY OF CERAMICS -- L. L. Snead. D. P. White, S. J. ankle (Oak Ridge National Laboratory), J. Oconor, G. Hartsough and J. Errante (Brookhaven National Laboratory). 457

An in-situ DC conductivity capsule has been constructed and tested at the High **Flux** Beam Reactor (HFBR) at Brookhaven National Laboratory. The prototype capsule contained two alumina and one silicon nitride polycrystalline samples. In-situ measurement of the DC conductivity was made utilizing custom-made mineral insulated triaxial cables with a guard ring configuration for the samples **and** the sample holders. Sample and capsule temperatures were also measured in-situ. **Thii** prototype capsule was designed to operate at ITER relevant temperatures of 80°C (alumina) and 300°C (alumina and silicon nitride) with an applied electric field of 133 V/mm. It is significant that a large prompt increase in conductivity occurred but that no permanent radiation induced electrical degradation was observed in alumina after irradiation at 80°C ***to*** a neutron dose of >0.1 dpa

8.10 THE EFFECT OF IONIZING AND DISPLACIVE RADIATION ON THE THERMAL CONDUCTIVITY OF ALUMINA AT LOW TEMPERATURE—D. P. White (Oak Ridge National Laboratory). 467

Microwave heating of plasmas in fusion reactors requires the development of windows through which the microwaves can pass without great losses. The degradation of the thermal conductivity of alumina in a radiation environment is an important consideration in reliability studies of these microwave windows. Several recent publications have calculated the radiation induced degradation at high temperature and at low temperature. The current paper extends the low temperature calculations in order to determine the effect of phonon scattering by radiation produced vacancies and by conduction hand electrons due ***to*** RIC on the thermal conductivity at 77K. These low temperature calculations are of interest because the successful application of high power (>1 MW) windows for electron cyclotron heating systems in fusion reactors will most likely require cryogenic cooling to take advantage of the low loss tangent and higher thermal conductivity of candidate window materials at these temperatures.

It is found that vacancy scattering can significantly reduce the thermal conductivity; for example, a vacancy concentration of 0.01 per atom leads to a fractional change in thermal conductivity of about 90%. It is also concluded that the scattering of phonons by electrons in the conduction band due to RIC does not lead to **a** large reduction in the thermal conductivity.

8.11 MICROSTRUCTURE OF ION IRRADIATED CERAMIC WSULATORS -- S. J. Zinkle (Oak Ridge National Laboratory). 475

Cross-section transmission electron microscopy was utilized to examine the radiation-induced microstructural changes in Al_2O_3, MgO, Si_3N_4 and $MgAl_2O_4$ after irradiation with a wide variety of energetic ion beams. The relative influence of ionizing and displacive radiation was studied by systematically varying the mass and energy of the. bombarding ions between **1** MeV H^+ and **4** MeV Zr^+ ions. The measured ion **ranges** were between 1 and 15% greater than the ranges calculated by the TRIM code, with the largest discrepancies occurring for intermediate mass ions. The implanted ions exerted a strong influence on the overall microstructural evolution of the irradiated ceramics. Numerous microstructural features (e.g., amorphization, colloids) were produced in the implanted ion regions which could not be produced in irradiated regions that were well separated from the implanted ions. The microstructural evolution in regions well separated from the implanted ions was found to depend strongly on the mass and energy of the bombarding ion. Light ion irradiation produced a significant enhancement in point defect diffusion (evidenced by defect-free zones at

the surface and adjacent to internal defect sinks) compared to heavy ion irradiation at the same damage rate. Similarly, irradiation with a given ion at a higher flux generally produced an increased amount of ohservahle diffusion. In some cases such as 1 MeV proton irradiation, observable defect clusters did not form. This suppression in defect cluster formation is attributed to the high amount of point defect recombination associated with ionization enhanced diffusion that occurs during energetic light ion irradiation.

8.12 ISSUES RELATED TO MECHANICAL PROPERTIES OF NEUTRON-IRRADIATED CERAMICS -- F. W. Clinard, Jr. (Los Alamos National Laboratory), W. Dienst (KfK Karlsruhe), and E. H. Famum (Los Alamos National Laboratory). 493

Ceramics will he used for a numher of applications in fusion devices, where their dielectric propertics, high strength, refractoriness. or low activation characteristics are required. In all cases starting mechanical properties must he adequate to preserve structural integrity, and the material must tolerate irradiation damage without undue degradation. A review of the irradiation response of four prototypical candidate ceramics shows that significant deterioration of mechanical properties can he expected for most applications in ITER. Future studies are identified that will establish the basis for specifying developing ceramics that can meet the needs of ITER and subsequent fusion reactor concepts.

8.13 DIMENSIONAL STABILITY, OPTICAL AND ELASTIC PROPERTIES OF $MgAl_2O_4$ SPINEL IRRADIATED IN FFTF TO VERY HIGH EXPOSURES--F. A. Gamer, G. W. Hollenberg and J. L. Ryan (Pacific Northwest Laboratory), Z. Li (Argonne National Laboratory), C. A. Black and R. C. Bradt (University of Nevada-Keno). 502

Stoichiometric $MgAl_2O_4$ spinel specimens irradiated in FFTF-MOTA at temperatures between 385 and 750°C to fluences ranging from 2.2 to 24.9 x 10^{22} n cm^{-2} (E>0.1 MeV) darken significantly, hut do not develop any significant loss in weight or change in dimensions. Similar behavior was observed in both single crystal and fully dense polycrystalline specimens. Measurements of elastic constants hy an ultrasonic technique show lhat no measurable changes occur as a result of the irradiation. These and other results confirm the stability of this material for fusion application as an electrical insulator.

8.14 INFLUENCE OF HIGH DOSE NEUTRON IRRADIATION AT 385 AND 750°C ON THE MICROHARDNESS OF $MgAl_2O_4$ SPINEL--C. A. Black (University of Nevada-Reno), F. A. Garner (Pacific Northwest Laboratory. R. C. Bradt (University of Nevada-Reno). 508

High-purity specimens of stoichiometric $MgAl_2O_4$ single crystal spinel and a hot-pressed polycrystalline cerrunic spinel were irradiated to exposures as large as 24.9 x 10^{22} cm^{-2} (E>0.1 MeV) in FFTF at 385°C and 750°C. The specimens did not develop any brittleness or fragility, and maintained their physical integrity. Microhardness measuremenls revealed that initially all specimens hardened a small amount and then recovered slightly. At the lower irradiation temperature, the dependence of microhardness on orientation Observed prior to irradiation tended to disappear. There was also some evidence that a secondary slip system was being activated. Following 750°C irradiation, the orientation dependence was not lost, and the evidence for activation of a secondary slip system was stronger.