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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, JPlO. 11, 13, 16 accumulated 18 dpa and were removed from the reactor in September 1991. JP14 was removed from the reactor at the end of cycle 310 (September 1992) after achieving a peak dose of 34 dpa Capsules JP9, 12, and 15 completed 35 cycles on April 1, 1994 and were removed from the target region. Each capsule had accumulated a peak dose of approximately 57 dpa.

Three new capsules, HFIR-MFE-JP20, 21, and 22, have been designed to complete the original experiment matrix of the JP9 through JP16 series capsules. Irradiation began in December 1993 with the start of cycle 322. As of April 1,1994 (end of cycle 324), each capsule had achieved approximately 4.9 dpa.

The RB* experiments are the continuation of the ORR spectrally tailored experiments. In this stage, a hafnium liner is used to modify the spectrum to simulate fusion helium to displacements per atom (Hddpa) ratio in the austenitic steel specimens. Of the four capsules in this set, two (60J-1 and 330J-1) were disassembled during this reporting period after accumulating a peak dose in HFIR of 11 dpa Two capsules (200J-1 and 400J-1) are in the reactor and have achieved a dose of 5.5 dpa

1.2 THE COBRA-1B IRRADIATION EXPERIMENT IN EBR-II – H. Tsai, A. G. Hins, R. V. Strain, and D. L. Smith (Argonne National Laboratory). 8

Design of the COBRA-1B irradiation experiment began in this reporting period and is in progress. The target reactor insertion date for COBRA-1B is September 1994. Technical and programmatic feasibility approval for the experiment has been granted by EBR-II Operations.

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2.1 REVISED NEUTRON DOSIMETRY RESULTS FOR THE MOTA-2A EXPERIMENT IN FFTF – L. R. Greenwood (Pacific Northwest Laboratory). 13

Revised neutron flnence and damage values are reported for the MOTA-2A experiment in the Fast Flux Test Facility (FFTF). This revision corrects an error with processing of the $^{235}U(n,f)$ reaction. Net corrections are on the order of 5%.

2.2 REANALYSIS OF NEUTRON DOSIMETRY FOR THE MOTA-1A/1B EXPERIMENTS IN FFTF – L. R. Greenwood (pacific Northwest Laboratory). 15

Neutron fluence and spectral measurements and radiation damage calculations have been revised for the MOTA-1A, 1B experiments in the Fast Flux Test Facility (FFTF). The data were revised in order to correct for nuclear burnup effects for several of the radiometric monitors, thereby inneasing the fluence and damage up to 20%, according to position the MOTA assembly. The MOTA-1A experiment was irradiated from January 18,1983, to October 23,

1983, for an exposure of 202.0 EFPD. MOTA-1B was irradiated from January 1,1984. to April 23,1984, for an exposure of 109.4 EFPD. Neutron fluence and dpa values are presented for both assemblies.

2.3. ** R ODOSIMETRY FOR THE MOTA-2B EXPERIMENT IN FFTF – L. R. Greenwood and L. S. Kellogg (Pacific Northwest Laboratory). 21

Neutron fluence and spectral measurements and radiation damage calculations are reported for the Materials Open Test Assembly (MOTA-)2B experiment in the Fast Flux Test Facility (FFTF). The irradiation was conducted from May 27,1991, to March 19,1992, for a total exposure of 203.3 EFPD (effective full-power days). The maximum fluence was about 9 x 10^{22} n/cm^2, 6 x 10^{22} above 0.1 MeV producing about 24 dpa in iron. Neutron fluence and radiation damage maps are presented for the entire MOTA assembly based on the analysis of ten neutron spectral measurements and fourteen additional flux gradient measurements.

2.4 ** R O DOSIMETRY FOR THE MOTA-IC EXPERIMENT IN FFTF – L. R. Greenwood (Pacific Northwest Laboratory). 28

Neutron fluence and spectral measurements and radiation damage calculations have been completed for the Materials Open Test Assembly (MOTA)-1C experiment in the Fast Flux Test Facility (FFTF). Although this irradiation was completed in 1985. the neutron dosimetry data were never analyzed. The MOTA-1C experiment was irradiated in cycles 5 and 6 from June 16, 1984, to June 24, 1985, for an exposure of 256.7 EFPD (effective full power days) The irradiation was conducted at 400 MW. Neutron fluence and dpa dues are presented.

2.5 TRANSMUTATIONS OF ELEMENTS UNDER IRRADIATION AND ITS IMPACT ON ALLOYS COMPOSITION – I. C. Gomes and D. L. Smith (Argonne National Laboratory). 33

This study presents a comparison of nuclear transmuration rates for candidate fusion first wall/blanket structural materials in available fission test reactors with those produced in a typical fusion spectrum. The materials analyzed in this study include a vanadium alloy (V-4Cr-4Ti), a reduced activation martensitic steel (Fe-9Cr-2WVTa), a high conductivity copper alloy (Cu-Cr-Zr), and the SiC compound The fission irradiation facilities considered include the EBR-II (Experimental Breeder Reactor) fast reactor, and two high flux mixed spectrum reactors, HFIR (High Flux Irradiation Reactor) and SM-3 (Russian reactor). The transmutation and dpa rates that occur in these test reactors are compared with the calculated transmutation and dpa rates characteristic of D-T fusion first wall spectrum. In general past work has shown that the displacement damage produced in these fission reactors can be correlated to displacement damage in a fusion spectrum; however, the generation of helium and hydrogen through threshold reactions [(n,xα) and (N,xp)] are much higher in a fusion spectrum. As shown in this study, the compositional changes for several candidate structural materials exposed to a fast fission reactor spectrum are very low, similar to those for a characteristic fusion spectrum. However, the relatively high thermalized spectrum of a mixed spectrum reactor produces transmutation rates quite different from the ones predicted for a fusion reactor, resulting in substantial differences in the fmal composition of several candidate alloys after relatively short irradiation time. As examples, the transmutation rates of W, Ta, V, Cu, among others. differ considerably when the irradiation is performed under a mixed spectrum reactor's and fusion fvst wall's spectrum. The out-of-core positions in mixed spectrum reactors can be partially shielded against low energy neutrons, *e.g.*, by hafnium. to reduce (be transmuration rates but the displacement rates at these positions are much lower than those for the core positions. Fast reactors (EBR-II) provide the only possibility for obtaining high damage rates without producing significant compositional effects in vanadium alloys, ferritic steels, and copper alloys.

2.6 CALCULATIONS TO DETERMINE THE FEASIBILITY OF USING A BORON CARBIDE SHIELD FOR SPECI'RAL TAILORING IN THE HFIR FOR VANADIUM ALLOY IRRADIATION–R. A. Lillie (Oak Ridge National Laboratory). 43

A neutron transport calculation has been performed to determine the feasibility of creating a shielded facility in the Be reflector of the HFIR. It was found that the B_4C provides satisfactory spectral tailoring to reduce production of Cr below the tolerance limit for Cr in the original alloys. The shield would have to be replaced after eight cycles or 5 dpa, but the Operations schedule for the HFIR makes it easier to replace the shield after seven cycles. Three shields would then have to be used in order to reach 10 dpa. This facility would provide the equivalent of a low-temperature fast neutron position in the HFIR for development and preliminary qualification of vanadium alloys for fusion devices.

2.1 TRANSMUTATION IN COPPER-BASE BRAZING MATERIALS – F. A. Gamer, L. R. Greenwood, and D. J. Edwards (Pacific Northwest Laboratory). 41

Brazing compounds containing silver or gold will transmute during irradiation, forming significant amounts of either cadmium or mercury, respectively, in all neutron spectra of current interest to the fusion materials program. The impact of this transmutation on braze performance and on development of fssion-fusion correlations has yet to be determined. but will be addressed in an irradiation experiment that was recently completed in FFTF-MOTA. A lesser amount of transmutation will occur in brazes containing tin, but no specimens containing brazes with tin have yet been irradiated.

3.0 MATERIALS ENGINEERING AND DESIGN REQUIREMENTS. 51

ITER MATERIALS PROPERTIES HANDBOOK – J. W. Davis, McDonnell Douglas Aerospace. 53

In December the ITER Joint Central Team authorized the creation of a Materials Properties Handbook. This Handbook is a cooperative activity between the four parties of ITER (Japan, the E.U.. the RF., and the U.S.). The U.S. Home Team has been selected to coordinate the documentation, while all four Parties will provide data During this period the effort focused on organizing this activity, developing a format for use by all of the parties, and demonstrating the ability to electronically transfer files between the four parties. First draft of the Handbook is scheduled for July.

4.0 FUNDAMENTAL MECHANICAL BEHAVIOR.. 57

No contributions.

5.0 RADIATION EFFECTS, MECHANISTIC STUDIES, THEORY, AND MODELING. 59

5.1 VOID SWELLING IN BINARY FE-CR ALLOYS AT 200 DPA – D. S. Gelles (Pacific Northwest Laboratory). 61

Microstructural examinations have been performed on a series of binary Fe-Cr alloys irradiated in the FFTF/MOTA at 425°C to 200 dpa. The data represent the highest swelling levels reported in neutron irradiated ferritic alloys. The alloy compositions ranged from 3 to 18%Cr in 3% Cr increments. and the irradiation temperature corresponded to the peak swellmg condition for this alloy class. Density measurements showed swelling levels as high as 7.4%, with the highest swelling found in the Fe9Cr and Fe6Cr alloys. Microstructural examinations revealed that the highest swelling conditions contained well-developed voids, often as large as 100 nm, and a dislocation network comprised of both $\frac{a}{2}$<111> and a<100> Burgers vectors. Swelling was lower in the other alloys, and the swelling reduction could be correlated with

increased precipitation. These results are considered in light of the current theories for low swelling in femtic alloys, but no theory is available to completely explain the results.

5.2 DATA BASE ON PERMEATION, DIFFUSION, AND CONCENTRATION OF HYDROGEN ISOTOPES IN FUSION REACTOR MATERIALS – J. L. Brimhall, E. P. Simonen, and R. H. Jones (Pacific Northwest Laboratory). 80

In evaluating fusion reactor performance. knowledge about permeation. diffusion, and concentration of hydrogen isotopes throughout the structure is of critical importance. The data base on hydrogen isotope permeation in relevant fusion reactor materials is reviewed in this report. Comparisons are made within alloys of the same type as well as with all the other classes of alloys. Both gas-driven permeation (GDP) and plasma-driven permeation (PDP) are included. Data on GDP behavior in Fe and Ni alloys arc relatively consistent. In metals that have a high solubility and/or are hydride formers, e.g., V. Nb, Ti, there is much more variation in the data. and surface effects play a very dominant role. Permeation under plasma conditions is less well understood as more variables enter the relationship. e.g., re-emission phenomena, internal concentration gradients, radiation effects, etc. The data show that materials do not necessarily rank in the Same order under PDP conditions as under GDP conditions. A summary of the factors that influence permeation and the relative magnitude of their effect is given in the report.

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6.1 FERRITIC MARTENSITIC AND BAINITIC STEELS 109

6.1.1 DEVELOPMENT OF LOW-CHROMIUM, CHROMIUM-TUNGSTEN STEELS FOR FUSION – K. L. Klueh and D. J. Alexander (Oak Ridge National Laboratory). 111

Iligh-chromium (9-12% Cr) Cr-Mo and Cr-W ferritic steels are favored as candidates for fusion applications. In the early work to develop reduced-activation steels, an Fe-2.25Cr-2W-0.25V-0.1C steel (designated 2 1/4Cr-2WV) had better strength than an Fe-9Cr-2W-0.25V-0.07Ta-0.1C (9Cr-2WVTa) steel (compositions are in weight percent). However, the 2 1/4Cr-2WV bad less-than-adequate impact toughness. as determined by the ductile-brittle transition temperature and upper-shelf energy of a Charpy impact test. Because low-chromium steels have some advantages over high-chromium steels. a program to develop low-chromium steels is in progress. Microstructural analysis indicated that the reason for the inferior impact toughness of the 2 1/4Cr-2WV was the granular-bainite microstructure obtained when the steel was normalized. Properties *can* be improved by developing an acicular bainite structure by increasing the cooling rate after austenitization. Alternatively. acicular bainite can be promoted by increasing the hardenability. Hardenability was changed by adding *small* amounts of boron and chromium to the 2 1/4Cr-2WV composition. A combination of B, Cr, and Ta additions resulted in low-chromium reduced-activation steels with mechanical propenies comparable to those of 9Cr-2WVTa.

6.1.2 MICROSTRUCTURE-MECHANICAL PROPERTIES CORRELATION OF IRRADIATED CONVENTIONAL AND REDUCED-ACTIVATION MARTENSITIC STEELS – R. L. Klueh (Oak Ridge National Laboratory). Ji-Jung Kai (National Tsing Hua University, Taiwan). and D. J. Alexander (Oak Ridge National Laboratory). 12X

Tensile, Charpy, and transmission electron microscopy specimens of two conventional steels. modified 9Cr-1Mo (9Cr-1MoVNb) and Sandvik HT9 (12Cr-1MoVW), and two reduced-activation steels, Fe-9Cr-2W-0.25V-0.1C (9Cr-2WV) and Fe-9Cr-2W-0.25V-0.07Ta-0.1C (9Cr-2WVTa), were irradiated in the Fast Flux Test Facility. Before irradiation, $M_{23}C_6$ was the primary precipitate io all four steels, which also contained some MC. Neutron irradiation did not substantially alter the $M_{23}C_6$ and MC. No new phases formed during irradiation of the 9Cr-2WV and 9Cr-2WVTa, but chi-phase precipitated in the 9Cr-1MoVNb and chi-phase

and α' precipitated in the 12Cr-1MoVW. Irradiation-produced dislocation loops were observed in 9Cr-2WV, 9Cr-2WVTa, and 12Cr-1MoVW. The irradiation-produced microstructural changes caused the steels to harden, as measured by the change in yield stress. Hardening was correlated with a change in the Charpy impact properties of 9Cr-1MoVNb, 12Cr-1MoVW, and 9Cr-2WV. Although irradiation caused a yield stress increase of the 9Cr-2WVTa similar to that for 9Cr-2WV and 9Cr-1MoVNb, the change in Charpy properties was considerably less for 9Cr-2WVTa. This difference in Charpy behavior of the 9Cr-2WVTa with that of the 9Cr-2WV and 9Cr-1MoVNb was amibuted to differences in the fracture stress-temperature relationship and/or the flow stress-temperature relationship between the 9Cr-2WVTa and the other two 9Cr steels.

6.1.3 CONSEQUENCES OF NO POSTWELD HEAT TREATMENT ON THE FRACTURE TOUGHNESS OF LOW ACTIVATION MARTENSITIC STEELS – Huaxin Li (Associated Western Universities, Northwest Division), D. S. Gelles and R. H. Jones (Pacific Northwest Laboratory). 142

Fracture toughness has been measured for two conditions of low activation martensitic steel F-82H, the first, a fully tempered condition and the second, completely untempered. In the fully tempered conditions, F-82H has a J_{IC} toughness of 284 kJ/m^2 (K_{IC} = 263.8 MPa$\sqrt{m}$), but in the untempered condition, the J_{IC} toughness is 18.4 kJ/m^2 (K_{IC} = 64 MPa$\sqrt{m}$). The consequences of not tempering are discussed in light of these measurements.

6.1.4 MICROSTRUCTURAL EXAMINATION OF COMMERCIAL FERRITIC ALLOYS AT 200 DPA – D. S. Gelles (Pacific Northwest Laboratory). 146

Microstructures and density change measurements are reported for martensitic commercial steels HT9 and modified 9Cr-1Mo (T9) and oxide dispersion strengthened ferritic alloys MA956 and MA957 following irradiation in the FFTF/MOTA at 420°C to 200 dpa Swelling as determined by density change remains below 2% for all conditions. Microstructures are. found to be stable except in recrystallized grains of MA957, which are fabrication artifacts, with only minor swelling in the martensitic steels and α' precipitation in alloys with 12% or more chromium. These results further demonstrate the high swelling resistance and microstructural stability of the ferritic alloy class.

6.1.5 EFFECT OF HYDROGEN ON THE FRACTURE TOUGHNESS OF A FERRITIC/MARTENSITIC STAINLESS STEEL – H. Li (Associated Western Universities-Northwest Division), R. H. Jones and D. S. Gelles (Pacific Northwest Laboratory), and J. P. Hirth (Washington State University-Pullman). 161

Effects of hydrogen (H) on the mixed-mode I/III critical J integrals (J_{MC}) and tearing moduli (T_M) were examined for a ferritic/martensitic 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. A specimen with 0 degree crack angle is the same as a standard mode I compact tension specimen. Specimens were heat-treated at 1000°C/20 h/air cooled (AC)-1100°C/7 min/AC-700°C/2 h/AC. The specimens were charged with H at a hydrogen gas pressure of 138 MPa at 300°C for two weeks. which resulted in a H content of 6 ppm(wt), as measured with the "inert gas fusion" technique. J_{MC} and T_M values were determined with the single specimen technique. Crack lengths were calculated by means of partial unloading compliances. The preliminary results from J integral tests showed that introducing H decreased J_{MC} and T_M values as compared to those without H. However, the presence of H did not change the dependence of J_{MC} and T_M values on the crack angles. Both mode I J and T values (J_{IC} and TI) exhibited the highest values. Both the minimum J_{MC} and

T_M values occurred at a crack angle between 35 and 55 degrees, corresponding to the load ration (σ_{iii}/σ_i) of 0.7 to 1.4.

6.2 AUSTENITIC STAINLESS STEELS. 171

6.2.1 FRACTURE TOUGHNESS OF IRRADIATED CANDIDATE MATERIALS FOR ITER FIRST WALL/BLANKET STRUCTURES – D. J. Alexander, J. E. Pawel, M. L. Grossbeck, A. F. Rowcliffe (Oak Ridge National Laboratory). and K. Shiba (Japan Atomic Energy Research Institute). 173

Candidate materials for first wall/blanket structures in ITER have been irradiated to damage levels of about 3 dpa at nominal irradiation temperatures of either 90 or 250°C. These specimens have been tested over a temperature range from 20 to 250°C. The results show that irradiation at these4 temperatures reduces the fracture toughness of austenitic stainless steels, but the toughness remains quite high. The toughness decreases as the test t e m p t u r e increases. Irradiation at 250°C is more damaging than at 90°C, causing larger decreases in the fracture toughness. Ferritic steels are embrittled by the irradiation, and show their lowest toughness at room temperature.

6.2.2 LOW TEMPERATURE RADIATION-INDUCED SEGREGATION RELATIVE TO IASCC AND ITER – E. P. Simonen. R. H. Jones, L. A. Charlot, and S. M. Bruemmer (Pacific Northwest Laboratory). 194

Radiation induced segregation at grain boundaries in stainless steel has been evaluated as a function of temperature. Segregation caused by ion irradiation was measured and compared to temperature-dependent model predictions. The model was subsequently used to extrapolate predictions to ITER operating conditions. For ion irradiation, segregation was found to extend to temperatures below 400°C which implies that segregation is expected at temperatures less than 200°C for ITER irradiation damage rates. The model predictions were in accord with Cr depletion levels and profile widths as measured using FEG-STEM examination of ion irradiated grain boundaries. The model was further shown to be consistent with measured depletion levels at grain boundaries in 316 stainless steel irradiated in a light-water reactor at 288°C. The findings of this study suggest that chromium depletion and hence IASCC may be a concern for stainless steel structures irradiated in ITER high-temperature water environments.

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6.3.1 SWELLING AND STRUCTURE OF VANADIUM ALLOYS IRRADIATED IN THE DYNAMIC HELIUM CHARGING EXPERIMENT – H. M. Chung, B. A. Loomis, H. Tsai, L. Nowicki, J. Gazda, and D. L. Smith (Argonne National Laboratory). 204

Combined effects of dynamically charged He and neutron damage on density change, void distribution. and microstructural evolution of V-4Cr-4Ti have been determined after irradiation to 18-31 dpa at 425-600°C in the Dynamic Helium Charging Experiment (DHCE). For specimens irradiated to ≈18 dpa at 600°C with an estimated He/dpa ratio of 3-9 appm He/dpa (a range similar to the fusion-relevant ratio of ≈5 appm Hddpa). only a few microvoids were observed at the interface of the grain matrix and some Ti(O,N,C) precipitates. No microvoids were observed either in the grain matrix or near grain boundaries. It seems that most of the dynamically produced He atoms are trapped in the grain matrix without significant void nucleation or growth. In this group of DHCE specimens, density changes were similar to those of non-DHCE (negligible He generation) specimens for a comparable fluence and irradiation temperature. Significant numbers of microvoids in the grain matrix and on localized grain boundaries were present only in specimens irradiated to =31 dpa at 425°C in high-tritium capsules, i which an excessive amount of helium was generated (estimated Hddpa ratio of 10-35 appm). Discontinuous regions of void segregation on grain boundaries were

observed, and void number-density was significantly lower than in other alloys irradiated in tritium-trick experiments. The grain-boundary voids seem to have been produced by decay of tritium segregated to grain boundaries in the specimens at the onset of irradiation in high-tritium capsules (an artifact effect similar to tritium charging at 400°C in a tritium-hick experiment), rather than by segregation of He produced in the grain mahix. Density changes of the specimens were significantly higher than those from non-DHCE irradiations. Microstructural evolution in V-4Cr-4Ti was similar for DHCE and non-DHCE irradiations, except for void number-density and distribution. As in non-DHCE specimens, inadiation-induced precipitation of ultrafine Ti_5Si_3 was observed for irradiation at 600°C but not at 425°C.

6.3.2 EFFECT OF DYNAMICALLY CHARGED HELIUM ON MECHANICAL PROPERTIES OF V-4Cr-4Ti – H. M. Chung, B. A. Loomis. H. Tsai, L. Nowicki. D. E. Busch, and D. L. Smith (Argonne National Laboratory). 212

V-4Cr-4Ti has been reported to be virtually immune to irradiation embrittlement under conditions of negligible helium generation: DBTTs determined from one-third-size Charpy-impact specimens irradiated in a non-DHCE experiment were <-200°C. The effects of dynamically charged helium and tritium, and of neutron damage, on DBT of the alloy were determined after irradiation in the Dynamic Helium Charging Experiment (DHCE). TEM disks irradiated to 18-31 dpa at 425-600°C were fractured by repeated bending in a low-temperature bath. Miniature specimens were used because no Charpy-impact specimens were included in the DHCE experiment Ductile-brittle transition behavior was determined from quantitative SEM fractography. Brittle fracture surface morphology was not observed at >-120°C regardless of the level of dpa damage or helium and tritium content in the alloy. Predominantly brittle-cleavage fracture morphologies were observed only at -196°C in some specimens irradiated to 31 dpa at 425°C and in which the estimated helium and tritium contents were highest This indicates a DBlT between -175 and -200°C in specimens containing the highest levels of dpa damage, helium, and tritium in the DHCE experiment. No intergranular fracture was observed.

6.3.3 STATUS OF THE DYNAMIC HELIUM CHARGING EXPERIMENT – H. Tsai, R. V. Strain, H. M. Chung, and D. L. Smith (Argonne National Laboratory). 220

Seven DHCE capsules containing vanadium-alloy specimens were irradiated in the MOTA-2B vehicle in FFTF to =20 to 29 dpa. In this and the previous reporting period?., equipment and procedures were developed at Argonne National Laboratory-East to disassemble these capsules. Six of the seven capsules have been disassembled. (The seventh capsule was processed in April 1994. immediately following this reponing period.) Effluence of tritium into the cell exhaust was <2% of the total inventory. No contamination incidents occurred during the DHCE disassembly work. Initial testing of the rehieved specimens is under way.

6.3.4 FABRICATION OF VANADIUM ALLOY FORMS FOR SPECIMEN PREPARATION – M. L. Grossbeck, T. K. Roche, W. S. Eatherly, F. F. Dyer, and N. H. Rouse (Oak Ridge National Laboratory). 225

The alloy V–5Cr–5Ti (Teledyne Wah Chang heat 832394) was received in the form of a 6.35 mm plate and was rolled to 3.81 mm plate and 0.76 mm sheet The 3.81 mm plate was used for Charpy impact specimens and the 0.76 mm sheet was used for tensile specimens and welding experiments. Since interstitial impurities are important strengthening and embrittling agents in refractory metals, all anneals had to be &ne in ultra–high vacuum furnaces.

6.3.5 WELDING DEVELOPMENT FOR V-Cr-Ti ALLOYS – G. M. Goodwin and J. F. King (Oak Ridge National Laboratory). 235

Welds have been produced and characterized using the gas-tungsten arc (GTA) and electron beam (EB) welding processes. Thin sheet (0.75 mm) welds were made with three levels of interstitial contamination. and hardness and tensile properties were found to he strongly affected by oxygen pickup. Thick-section (6mm) welds have been produced using both processes, and no embrittlement is experienced when high purity atmosphere is maintained. Metallographic examination shows a narrow, but coarse grained, heat affected zone for the GTA welds. Transition joint welding development between vanadium alloy and stainless steel has shown encouraging results.

6.3.6 CHEMICAL AND MECHANICAL INTERACTIONS OF INTERSTITIALS IN V-5%Cr-5%Ti – J. H. DeVan, J. R. DiStefano, and J. W. Hendricks (Oak Ridge National Laboratory). 240

Gas-metal reaction studies of V-5Cr-5Ti were conducted to determine the kinetics of reactions with H_2 and O_2, respectively, at 400-500°C. Reaction rates were determined through weight change measurements and chemical analyses. and effects on mechanical properties were evaluated by room temperature tensile tests. Exposure to hydrogen at pressures between 10^{-4} to 10^{-2} torr lowered the tensile ductility in the case of coarse-grained specimens but had little effect on finer-grained specimens. Similarly oxygen uptake at 500°C, at concentrations as low as 200 ppm (by weight), significantly lowered the ductility of the coarser-grained but not the finer-grained material.

6.3.7 THE RELATIONSHIP BETWEEN RECRYSTALLIZATION TEMPERATURE, GRAIN SIZE. AND THE CHARPY IMPACT PROPERTIES OF V-Cr-Ti ALLOYS – M. L. Grossbeck. D. J. Alexander, and A. F. Rowcliffe. 244

It is shown that the blunt notch Charpy impact properties of a V-5Cr-5Ti alloy are extremely sensitive to the grain size distribution developed during the final annealing treatment. Completely ductile behavior down to liquid niuogen temperatures can be induced, for example, by a treatment at 950°C, rather than annealing at 1125°C. It is proposed that the resistance to irradiation-induced shift in DBTT reponed by Argonne National Laboratory (ANL) researchers for the V-4Cr-4Ti alloy is related primarily to the exceptionally fine-grained microstructure developed during the low temperature processing route adopted for this alloy.

6.3.8 KOOM TEMPERATURE ELASTIC PROPERTIES OF V-5Cr-5Ti – W. A. Simpson (Oak Ridge National laboratory). 258

Elastic moduli were measured for a specimen of the alloy V-5Cr-5Ti in the annealed condition using pulse-echo techniques. The value of Young's modulus was determined to be 125.6 GPa, the shear modulus was determined to be 45.9 GPa, and the Piosson's ratio to be 0.367.

6.3.9 THERMOPHYSICAL PROPERTIES OF V-5Cr-5Ti – W. D. Porter, K. B. Dinwiddie. and M. L. Grossbeck (Oak Ridge National laboratory). 260

Measurements of thermal expansion of V-5Cr-5Ti have been made from room temperature to 600°C, and specific heat and thermal conductivity have been measured over the range of 100 to 600°C. In each case, the results have been fitted to a polynomial expression to provide a convenient form for design studies.

6.3.10 ATOM PROBE FIELD ION MICROSCOPY CHARACTERIZATION OF VANADIUM-TITANIUM-CHROMIUM ALLOYS – M. K. Miller (Oak Ridge National Laboratory) 269

The atom probe field ion microscope was used to examine two V-Cr-Ti alloys (V-5Cr-5Ti and V-4Cr-4Ti) which have been shown to have very different fracture properties. The alloys were examined for clustering to detect the presence of small precipitates as well as to determine the distribution of solute elements in the matrix. The elements Ti. Cr, and Si were examined and no clustering of any of these three elements was found.

6.3.11 ANALYSIS OF GRAIN BOUNDARIES IN A V-5CR-5TI ALLOY USING AUGER ELECTRON SPECTROSCOPY – D. N. Braski and M. L. Grossbeck (Oak Ridge National Laboratory) 272

A V-5Cr-5Ti specimen was fractured in-situ below its DBTT and the fracture surfaces analyzed by Auger Electron Spectroscopy (AES). About 70% of the fresh fracture surfaces were intergranular while the remainder were characteristic of cleavage fracture. Sub-micron-sized particles appeared to cover most of the grain boundary surfaces along with a few larger particles. Only one of these larger particles, probably a Ti-V sulfide/phosphide combination, contained Cl, at a level of only a few at.%. On the other hand, S, which is known to cause embrittlement in many alloy systems, was detected on virtually all of the grain boundary surfaces.

6.3.12 FRACTURE TOUGHNESS OF V-5Cr-5Ti ALLOY AT ROOM TEMPERATURE AND 100°C – H. Li (Associated Western Universities-Northwest Division), R. H. Jones (Pacific Northwest Laboratories), and J. P. Hirth (Washington State University-Pullman) 279

The critical mixed-mode I/III fracture toughness, J-integrals (J_{MC}), at room temperature (RT) and 100°C were examined for a V-5Cr-5Ti alloy. Fracture toughness at 100°C was evaluated with a J-integral test and at RT with a K (the stress intensity factor) test. The 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 45 degrees from the load line. Crack angles used in this study were 0, 15, and 45 degrees. A specimen with 0 degree crack angle is the same as a standard mode I compact tension specimen. In this limit J_{MC} becomes J_{IC}. Specimens were annealed at 1125°C for 1 hour in a vacuum of 10^{-7} ton. J_{MC} and mixed-mode tearing moduli (T_M) were determined at 100°C with the single specimen technique. Crack lengths were calculated with partial unloading compliances. The J_{MC} values at RT were calculated from critical stress intensity factors (K_c). The results showed that at RT the V-5CR-5Ti alloy was brittle and experienced unstable crack growth with a mixture of intergranular, cleavage. and some microvoid coalescence (MVC) fracture while at 100°C it exhibited high fracture toughness and fractured with a mixture of MVC and intergranular failure. SEM investigation showed that some cleavage facets initiated at grain boundaries. The results suggest a low intergranular fracture strength and tendency towards cleavage fracture at room tempture. Preliminary data from Auger electron microscopy showed significant sulfur segregation on grain boundaries. The possible mechanism which might reduce intergranular fracture strength is discussed.

6.3.13 INFLUENCE OF TRANSMUTATION ON MICROSTRUCTURE, DENSITY CHANGE, AND EMBRITTLEMENT OF VANADIUM AND VANADIUM ALLOYS IRRADIATED IN HFIR – S. Ohnuki and H. Takahashi (Hokkaido University. Sapporo, Japan), K. Shiba and A. Hishinuma (Japan Atomic Energy Research Institute. Tokai, Japan), J. E. Pawel (Oak Ridge National Laboratory). and F. A. Gamer (Pacific Northwest Laboratory) 293

Addition of 1 at% nickel to vanadium and V-10Ti, followed by irradiation along with the nickel-free metals in HFIR to 2.3 x 10^{22} n cm^{-2}, E>1.0 MeV (corresponding to 17.7 dpa) at

400°C, bas been used to study the influence of helium on microstructural evolution and embrittlement. Approximately 15.3% of the vanadium transmuted to chromium in these alloys The –50 appm helium generated from the $^{58}Ni(N,\gamma)^{59}Ni(n,\alpha)^{56}Fe$ sequence was found to exert much less influence than either the nickel directly or the chromium formed by transmutation.

The V-10-Ti and V-10Ti-1Ni alloys developed an extreme fragility and broke into smaller pieces in response to minor physical insults during density measurements. A similar behavior was not observed in pure V or V-1Ni. Helium's role in determination of mechanical properties and embrittlement of vanadium alloys in HFIR is overshadowed by the influence of alloying elements such as titanium and chromium. Both elements have been shown to increase the DBTT rather rapidly in the region of 10% (CR + TI). Since Cr is produced by transmutatation of V, this is a possible mechanism for the embrittlement Large effects on the DBTT may have also resulted from uncontrolled accumulation of interstitial elements such as C. N, and O during irradiation.

6.3.14 INFLUENCE OF FLUX-SPECTRA DIFFERENCES ON TRANSMUTATION AND SWELLING OF VANADIUM *ALLOYS* – F. A. Garner and L. R. Greenwood (Pacific Northwest Laboratory). 305

The swelling of vanadium and its alloys is known to be sensitive t the level of chromium in the alloy. Chromium is also the major transmutation product during neutron irradiation, however, and the production rate of chromium is very sensitive to neutron spectra. It *appears* that an apparent dependence of void swelling in pure vanadium irradiated in FFTF may arise in part from the spectral difference that accompanies changes in displacement rate.

6.3.15 DEPENDENCE OF VANADIUM ALLOY DENSITY ON STARTING COMPOSITION TRANSMUTATION AND SEGREGATION – F. A. GARNER (Pacific Northwest Laboratory) and B. A. Loomis (Argonne National Laboratory). 309

The density of vanadium alloys is strongly dependent on solute identity and concentration, especially for additions of chromium and titanium. One consequence of this dependence is that radiation-induced density changes arise from segregation and precipitation. Transmutation of vanadium to chromium can also lead to significant increases in density and can therefore complicate somewhat the interpretation of swelling data derived using immersion density techniques.

6.4 COPPER ALLOYS 313

6.4.1 COPPER *ALLOYS* FOR HIGH HEAT FLUX STRUCTURE APPLICATIONS – S. J. Zinkle (*Oak* Ridge National Laboratory) and S. A. Fabritsiev (D. V. Efremov Scientific Research Institute of Electrophysical Apparatus). 314

The mechanical and physical properties of copper alloys are reviewed and compared with the requirements for high heat flux structural applications in fusion reactors. High heat flux structural materials must possess a combination of high thermal conductivity and high mechanical strength. The three most promising copper alloys at the present time are oxide dispersion-strengthened copper ($Cu-Al_2O_3$) and two precipitation-hardened copper alloys (Cu-Cr-Zr and Cu-Ni-Be). These three alloys are capable of room temperature yield strengths > 400 MPa and thermal conductivities up to 350 W/m-K. All of these alloys require extensive cold working to achieve their optimum strength. Precipitation-hardened copper alloys such as Cu-Cr-Zr are susceptible to softening due to precipitate overaging and recrystallization during brazing, whereas the dislocation structure in $Cu-Al_2O_3$ remains stabilized during typical high temperature brazing cycles. All three alloys exhibit good resistance to irradiation-induced softening and void swelling at temperatures below 300°C. The precipitation-strengthened

alloys typically soften during neutron irradiation at temperatures above about 300°C and therefore should only be considered for applications operating at temperatures <300°C. Dispersion-strengthened copper may be used up to temperatures in excess of 500°C. Based on the available data, dispersion-strengthened copper ($Cu-Al_2O_3$) is considered to be the best candidate for high heat flux structural applications.

6.4.2 ROOM TEMPERATURE FATIGUE BEHAVIOR OF OFHC COPPER AND CuAl25 SPECIMENS OF TWO SIZES – A. Singhal and J. F. Stubbins (University of Illinois), B. N. Singh (Ria National Laboratory), and F. A. Gamer (Pacific Northwest Laboratory). . 342

Copper and its alloys are appealing for application in fusion reactor systems for high heat flux components where high thermal conductivities are critical, for instance, in divertor components. The thermal and mechanical loading of such components will be, at least in part, cyclic in nature. thus requiring an understanding of their fatigue behavior. This report describes the room temperature fatigue behavior of unirradiated OFHC (oxygen free high conductivity) copper and CuAl25 (copper strengthened with a 0.25% atom fraction dispersion of alumina). The response of two fatigue specimen sizes to strain controlled fatigue loading is examined, and differences in behavior are discussed. Specimens with the smaller size are not being irradiated in several reactors.

6.4.3 PREPARATION OF RUSSIAN COPPER ALLOY CREEP SPECIMENS – C. R. Eiholzer (Westinghouse Hanford Company), M. L. Hamilton (Pacific Northwest Laboratory), and V. Barabash (D. V. Efremov Institute). 351

Eleven creep specimens of the Russian copper alloy MAGT-0.2 were pressurized and welded closed. The specimens were pressurized to levels ranging from 7.2 to 15.6 MPa (1052 to 2267 psia). These pressures will produce hoop stresses between 55 and 185 MPA (7.977 to 26.830 psia) when irradiat 120°C.

6.4.4 VOID SWELLING OF PURE COPPER, CU-5NI AND CU-5MN ALLOYS IRRADIATED WITH FAST NEUTRONS – H. Watanabe (Kyushu University) and F. A. Gamer (Pacific Northwest Laboratory). 356

The effects of cold-work level and solute addition (nickel or manganese) on pure copper under fast neutron irradiation have been investigated. Neutron irradiation was conducted in the Fast Flux Test Facility (FFTF) at temperatures from 638 to 873K to doses ranging from 8 to 98 dpa In pure copper, the void swelling behavior was investigated as a function of cold work level. At 638 and 703K, 10% cold-work reduced swelling somewhat with little influence at higher cold-work levels. In comparison with pure copper, void swelling was suppressed by 5% nickel addition at 648K but slightly increased at 696K. Cu-5Mn resisted swelling in both annealed and 40% cold-worked specimens in all temptures examined. At 684K, void formation was not detected. Only stacking fault tetrahedra were observed in this specimen.

6.4.5 INFLUENCE OF NICKEL AND BERYLLIUM CONTENT ON SWELLING OF COPPER IRRADIATED IN COBRA-IA – F. A. Gamer (Pacific Northwest Laboratory) and B. N. Singh (Ria National Laboratory). 364

A series of annealed Cu-Ni dilute binary alloys were irradiated in COBRA-1A in EBR-II at temperatures of 373,420. and 500°C. Density measurements have been performed for specimens irradiated at 373°C to 1.78 x 10^{22} n cm^{-2} (–11.3 dpa for pure copper). At this temperature and dose level nickel additions suppress swelling. There is a possible indication that some type of segregation-related phenomenon also occurs, yielding radiation-induced densification.

6.5 ENVIRONMENTAL EFFECTS IN STRUCTURAL MATERIALS. 369

6.5.1 FABRICATION OF ALUMINUM NITRIDE AND ITS STABILITY IN LIQUID ALKALI METALS – K. Natesan and D. L. Rink (Argonne National Laboratory). 371

AlN has been selected as a prime candidate for electrically insulating the V-alloy first wall in the self-cooled ITER concept Several methods are being evaluated for fabricating coatings that have adequate thickness and desirable physical, electrical, chemical, and mechanical properties. Coatings developed thus far are being exposed to liquid Li at 350 and 400°C.

6.5.2 DEVELOPMENT OF ELECTRICAL INSULATOR COATINGS: IN-SITU ELECTRICAL RESISTANCE MEASUREMENTS ON CaO-COATED V-5%Cr-5%Ti IN LIQUID LITHIUM – J.-H. Park, G. Dragel, and R. W. Clark (Argonne National Laboratory). 376

The electrical resistance of CaO coatings produced on V-5%Cr-5%Ti by exposure of the alloy to liquid Li that contained 4 at% dissolved Ca was measured as a function of time at temperatures of 250 to 698°C. The solute element, Ca in liquid Li. reacted with the alloy substrate at 420°C to produce a CaO coating. The resistance of the coating layer was ≈0.4 and 35.7 Ω at 267 and 698°C, respectively. Thermal cycling between 267 and 698°C changed the resistance of the coating layer, which followed insulator behavior. These results and those reported previously suggest that thin homogeneous coatings can be produced on variously shaped surfaces by controlling the exposure time. temperature, and composition of the liquid metal. This coating method is applicable to reactor components of various shapes (e.g., inside/outside of tubes, complex geometrical shapes) because the coating is formed by liquid-phase reaction. The liquid metal can be used over and over because only the solutes are consumed within the liquid metal.

7.0 SOLID BREEDING MATERIALS AND BERYLLIUM. 383

7.1 PERFORMANCE OF CERAMIC BREEDER MATERIALS IN THE SIBELIUS EXPERIMENT – J. P. Kopasz and C. E. Johnson (Argonne National Laboratory) and D. L. Baldwin (Pacific Northwest Laboratories). 385

Lithium containing ceramics are among the leading candidates for use as tritium breeding materials in a fusion reactor. An issue affecting both the safety and economics of the reactor is the tritium inventory. The SIBELIUS experiment was designed to examine material compatibility between different components of a breeder blanket and to examine the tritium inventory in the components of the blanket The tritium inventory in each of the ceramics was determined by measurements at end of life and were found to be quite low and in agreement with those determined in tests with no beryllium present The inventory increased in the order lithium zirconate < lithium oxide < lithium orthosilicate < lithium aluminate with the inventory in the zirconate less than 0.03 wppm at 550°C.

7.2 AB INITIO CALCULATIONS FOR HYDROGEN ADSORPTION ON LITHIUM OXIDE SURFACES – A. Sutjianto, S. W. Tam, L. Curtiss, and C. E. Johnson (Argonne National Laboratory) and R. Pandey (Michigan Technological University). 392

The study of hydrogen chemisorption on the (110) and (111) lithium oxide surfaces have teen investigated by means of ab initio Hartree-Fock calculations. In general, for the n-layer slab (where n = 2 and 3) of the neutral (110) and for the three-layer slab of the neutral (111) surfaces, there is no hydrogen chemisorption. The oxygen 2p band of the (110) surface is stabilized by the presence of more stacking sequences of layers which prevents charge transfer to the hydrogen. In the case of a neutral (110) layer due to low coordination of the anion, there is hydrogen chemisorption in the form of OH- and $Li^+H^-Li^+$ with the chemisorption energy of

0.93 eV. For the neutral (111) surface, there is no hydrogen chemisorption since the oxygen layer is screened by the cation layers hindering the charge transfer to the hydrogen.

7.3 IRRADIATION OF LITHIUM ZIRCONATE PEBBLE-BED IN BEATRIX-II, PHASE II – R. A. Verrall and J. D. Sullivan (AECL Research, Chalk River Laboratories), O. D. Slagle and G. W. Hollenberg (Pacific Northwest Laboratory), and T. Kurasawa (Japan Atomic Energy Research Institute). 406

BEATRIX-II was an in situ tritium recovery experiment that was designed to characterize the behavior of lithium ceramics irradiated to a high bumup. and to assess their suitability for use in a fusion reactor blanket This paper describes the results from a vented canister containing lithium enriched lithium zirconate spheres (85% ^{6}Li) that were irradiated to a burnup of 5.2% (total lithium) in a steep temperature profile–400°C edge, 1100°C center. The reference sweep gas was He-0.1%H_2, with systematic tests using alternate compositions: He-0.01%H_2 and pure He (maximum duration 8 days). Tritium recovery decreased slightly at lower H_2 concentrations. For example, the buildup of inventory during a 4-day test in pure He was 0.8 Ci which was approximately 6.5% of the tritium generated in the lithium zirconate during that period. The steadiness of the bed central temperature and the high tritium release rate, together with low moisture release rate, indicate good performance of the zirconate bed.

8.0 CERAMICS 417

8.1 FATIGUE CRACK GROWTH RATE (FCGR) BEHAVIOR OF NICALON/SiC COMPOSITES – N. Miriyala, P. K. Liaw. and C. J. McHargue (University of Tennessee), L. L. Snead (Oak Ridge National Laboratory), and D. K. Hsu and V. K. Saini (Iowa State University). 419

Ultrasonic measurements were carried out on the Nicalon/SiC composite specimens to correlate elastic moduli with percentage porosity in the in-plane as well as through-thickness directions. Compact type C(T) specimens were cyclically loaded to study the fatigue crack growth behavior. However, it was difficult to initiate the crack in the specimens.

8.2 TENSILE BEHAVIOR OF IRRADIATED SIC FIBERS – M. C. Osbome, Rensselaer Polytechnic Institute, L. L. Snead, Oak Ridge National Laboratory, and D. Steiner, Rensselaer Polytechnic Institute. 428

Tensile results are presented for low oxygen Nicalon fibers neutron irradiated at damage levels of 0.013 displacements per atom (dpa), 0.13 dpa and 0.32 dpa. Single fibers were tensile tested and analyzed, using Weibull statistics. for mean strength and distribution. Tensile modulus was also determined. Using a diffractometer, the fiber grain size and percent crystallinity were determined. The mean svength and modulus decreased by 20% but then increased with the highest damage level tested. Both grain size and crystallinity decreased as dose increased. These initial results of low level neutron irradiation of low oxygen Nicalon fibers exhibit no substantial degradation of the properties investigated. Therefore, continued research at higher doses is recommended.

8.3 TEMPERATURE DEPENDENCE OF THE SUBCRITICAL CRACK GROWTH RATE OF A SiC/SiC COMPOSITE – R. H. Jones and C. H. Henager. Jr. (Pacific Northwest Laboratory). 438

The long term stability of SiC/SiC composites is an issue regarding their use for fusion structural applications only because there is insufficient data to determine if there is really a problem. Crack growth measurements have been made over the lemperature range of 800 to 1100°C, in high-purity Ar, to help assess the long term stability of these materials. Stage II crack growth rates ranging from 2 x 10^{-9} m/s at 1100°C down to 3 x 10^{-11} m/s at 800°C were

found for a 2-D SiC/SiC composite reinforced with 40% Nicalon fibers. The stage II velocities obeyed an Arrhenius relationship over this temperature range with an activation energy of 235 kJ/mol. At a fusion relevant temperature of 800°C, the crack velocity will result in a crack growth of 1 mm/yr. More creep resistant fibers are expected to reduce this crack growth amount to less than 1 x 10^{-2} mm/yr at 800°C. This is not an excessive amount of crack extension for a material with a fracture toughness of about 20 MPa $m^{1/2}$.

8.4 DEFECT ENERGETICS CALCULATIONS OF SiC USING THREE REPRESENTATIVE EMPIRICAL POTENTIALS – H. Huang, J. Wong, and N. Gboniem (University of California), and M. Baskes (Sandia National Laboratory). 445

Defect energetics is a critical issue not only for irradiation damage but also for semiconductor applications. Calculations of defect energetics rely mainly on using empirical potentials, which are developed based on information from equilibrium crystal configurations and which may not be suitable for investigating defected configurations. Therefore, not all potentials give accurate enough results. This paper fust calibrates MEAM for silicon carbide. Calculations are then made for defect energetics of silicon carbide using the MEAM and two other representative empirical potentials. Results obtained with these potentials are compared with available fust principle calculations and experimental data. The results not only provide more information for defect formation and migration in silicon carbide, but also provide a critical assessment of the three empirical potentials. The results are analyzed in terms of construction and calibration of the empirical potentials.

8.5 PHENOMENOLOGICAL INELASTIC CONSTITUTIVE EQUATIONS FOR SiC AND SiC FIBERS UNDER IRRADIATION – A. El-Azab and N. M. Ghoniem (University of California-Los Angeles). 459

Experimental data on irradiation-induced dimensional changes and creep in β-SiC and SiC fibers is analyzed, with the objective of studying the constitutive behavior of these materials under high-temperature irradiation. The data analysis includes empirical representation of irradiation-induced dimensional changes in SiC matrix and SiC fibers as function of time and irradiation temperature. The analysis also includes formulation of simple scaling laws to extrapolate the existing data to fusion conditions on the basis of the physical mechanisms of radiation effects on crystalline solids. Inelastic constitutive equations are then developed for SCS-6 SiC fibers, Nicalon fibers. and CVD SiC. The effects of applied stress, temperature, and irradiation fields on the deformation behavior of this class of materials are simultaneously represented. Numerical results are presented for the relevant creep functions under the conditions of the fusion reactor (ARIES IV) first wall. The developed equations can be used in estimating the macro mechanical properties of Sic-Sic composite systems as well as in performing time-dependent micro mechanical analysis that is relevant to slow crack growth and fiber pull-out under fusion conditions.

8.6 TRITIUM TRANSPORT AND RETENTION IN SILICON CARBIDE – S. W. Tam, J. Kopasz. and C. E. Johnson (Argonne National Laboratory). 485

A theoretical framework describing trapping, diffusion, and desorption of tritium from vapor-deposited p-silicon carbide has been proposed and an analytical version of this model was derived. This analytical model was used to extract napping. diffusion, and desorption parameters from a limited part of the experimental data set The resulting parameter-free model was then used to test its predictive capability by calculating the time-dependent release behavior for the high temperature regimes. General agreement with the release data was found supporting the suitability of the present approach. Tritium trapping was found to be strong. Both bulk diffusion and surface desorption make comparable contributions to tritium release from the silicon carbide. Improvements beyond the present analytical model to include dynamical detrapping can be achieved with the proposed framework.

8.7 RADIATION-INDUCEDCHANGES IN ELECTRIC, DIELECTRIC, AND OPTICAL PROPERTIES OF CERAMICS – S. J. Zinkle (Oak Ridge National Laboratory). 493

The data base on radiation-induced changes in the electric and dielectric properties of ceramic insulators is reviewed Ionizing radiation causes a prompt increase in the electrical conductivity that is generally proportional to the ionizing dose rate. The precise magnitude of the radiation induced conductivity is sensitive to the detailed manufacturing process and impurity content. The conflicting evidence regarding the possibility of permanent radiation induced electrical degradation (RIED) is summarized. Further work is needed to define the physical mechanism and practical operating limits of the RIED phenomenon. Displacement damage associated with neutron irradiation produces significant increases in the room temperature dielectric loss tangent of oxide ceramics for fast neutron fluences above 1×10^{22} n/m^2 (~0.001 dpa). A brief summary of the effects of irradiation on the optical properties of ceramics and silica optical fibers is also presented.

8.8 NEUTRON DAMAGE TO DIAGNOSTIC MIRRORS – E. H. Famum and F. W. Clinard, Jr. (Los Alamos National Laboratory), S. P. Regan (Johns Hopkins University, Baltimore, MD), and B. Schunke (E T Joint Undertaking, Abingdon, Ox, UK). 505

Diagnostic systems for fusion reactors will require mirrors capable of reflecting electromagnetic radiation in the soft x-ray, near uv, visible, and ir wavelengths. These components will be exposed to significant fluences of fast neutrons during use. Mirrors made from alternating layers of low-Z and high-Z materials were irradiated to a fluence of 1.1×10^{23} n/m^2 at 270-300°C, and subsequently evaluated for structural changes and in some cases changes in optical properties. Short-wavelength mirrors retained their structural integrity while exhibiting slight changes in reflectance; some long-wavelength mirrors showed structural degradation, while other did not. These results are discussed in terms of materials damage effects and possibilities for improvement of mirror performance under severe o p t i n g conditions.

8.9 ELECTRICAL PROPERTIES OF Al_2O_3 DURING IRRADIATION WITH SPALLATION NEUTRONS – E. H. Famum and F. W. Clinard, Jr. (Los Alamo National Laboratory). 512

DC and AC conductivity of Al2O3 ceramics were measured at elevated temperatures during irradiation with neutrons and gamma rays at the Los Alamos Spallation Radiation Effects Facility. DC conductivity was increased at start of irradiation, but was subsequently reduced as displacement damage was accumulated. AC conductivity appeared to increase at high levels of damage. The observed dc behavior is attributed to excitation of electrons into the conduction hand by ionizing radiation, followed by charge trapping and recombination at damage sites. The ac behavior, which resembles RIED, is attributed to conductivity in the residual gas surrounding the samples. Surface conductivity, while not the source of the apparent RIED. nevertheless is of sufficient magnitude to be of concern for fusion applications.

8.10 A COMPARISON OF THE EFFECT OF RADIATION ON THE THERMAL CONDUCTIVITY OF SAPPHIRE AT LOW AND HIGH TEMPERATURE – D. P. White (Oak Ridge National Laboratory). 517

It has been proposed that sapphire be used in insulating feedthroughs and windows in radiofrequency and microwave heating systems in fusion reactors. The degradation of the thermal conductivity of sapphire with irradiation is a concern in the design of these windows. It has been pmposed to cool microwave windows to liquid nitrogen temperatures to take advantage of the thermal conductivity peak, which occurs at approximately 30 K in unirradiated sapphire. and the lower dielectric loss tangent at these temperatures. In this paper the methods used to calculate the thermal conductivity and the changes which occur with changes in point

defect concentration will be reviewed and the results of the low temperature calculations will be compared to the high temperature calculations.

It is found that vacancy scattering can significantly reduce the thermal conductivity over a wide temperature range; for example, a vacancy concentration of 0.01 per atom leads to a fractional change of about 90% at 77K versus 43% at 400 K. This reduction has significance for the design and placement of radio frequency and microwave windows in fusion reactors.

8.11 NEUTRON-INDUCED CHANGES IN OPTICAL PROPERTIES OF $MgAl_2O_4$ SPINEL – A. Ibarra (CIEMAT, Madrid, Spain) and F. A. Garner (Pacific Northwest Laboratory) 523

High purity $MGAL_2O_4$ spinel specimens irradiated in FFTF-MOTA to very high exposure have been examined by three techniques to determine changes in their optical properties. Significant changes were observed in optical absorption, photoluminescence, and radioluminescence.