CONTENTS

Neutron fluence and spectral measurements and radiation damage calculations are reponed for the MOTA-1G experiment in die Fast I-lux lest Facility (FFTF). The irradiation was conducted from January 4, 1990, to March 19, 1991, for a total exposure of 299.7 EFPD. The maximum flucnce was 15.8 x 10^{22} n/cm^2, 10.0 x 10^{22} above 0.1 MeV producing 43.0 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 seven additional flux gradient measurements.

V-B and V-5Cr-B alloys have hceii irradiated in FFTF to assess the impact of helium generation ratc on microsuuctural evolution of vanadium-base alloys. Using this method it is somewhat difficult to separate the various effects of the original boron as well as the lithium and helium products of the ^{10}B (n, a) reaction. Post-irradiation measurements of helium concentrations have been completed for two sets of V-B alloys, one irradiated in-core and one below core. It appears that the boron doping technique used to produce the various alloys was reasonably successful in reaching the desired helium/dpa variations.

New nuclear section libraries (known as USACT92) have been created for activation calculations. A point-wise file was created from merging the previous version of the activation library, the U.S. Nuclear Data Library (ENDF/B-VI), and the European Activation File (EAF-2). 175 and 99 multi-group versions were also created. All the data are available at the National Energy Research Supercomputer Center.

Tungsten has been used in a variety of low activation ferritic alloys and also in copper composite alloys, both currently being irradiated in various fusion materials experiments. It has been proposed as an armor material also. However. in a manner that is strongly dependent on neutron spectra, tungsten transmutes strongly to rhenium and then osmium. This adds significant complexity to the interpretation of data developed in one spectral environment hut intended for application to another environment

Calculations have been performed to estimate the ionizing and displacive irradiation fields that will occur in ceramics during irradiation in accelerators aiid fission and fusion reactors. A useful measure of the relative strength of ionizing vs. displacive radiation is the ratio of the absorbed ionizing dose to the displacement damage dose. which in the case of ion irradiation is equal to the ratio of the electronic stopping power to the nuclear stopping power. In ceramics such as Al_2O_3, this ratio is about 20 at a fusion reactor first wall, and has a typical value of about 100 in a fusion reactor blanket region and in mixed spectrum reactors such as HFIR. Particle accelerator sources typically have much higher ionizing to displacive radiation ratios, ranging from about 2000 for 1 MeV protons to >10,000 for 1 MeV electrons.

A new decay data library has been completed. The new library contains up-to-date decay information (half-lives, branching ratios, decay energies, γ's energies and intensities). Activation responses such as the air and water biological hazard potentials, the waste disposal rating, and the biological dose are also included in this library. Recently developed cross-section libraries have been acquired to be used together with the decay data library.

Activation calculations have been made for the austenitic steel 316SS, the ferritic alloys HT-9, the titanium alloy Ti6Al4V, and the vanadium alloy V5Cr5Ti in a liquid metal (Na) design suggested recently for ITER. The calculations show that the vanadium alloy has the minimum short and long-term radioactivity and BHP. It also has the minimum decay heat all the time. The titanium alloy has less radioactivity than the austenitic and the ferritic alloys. However, the decay heat of this alloy could exceed that of the conventional alloys.

No contributions

Proper assessment of the mechanical properties along interfaces of bonded structures currently used in many fusion reactor designs is essential to compare the different fabrication techniques. A Mechanical Properties Microprobe (MPM) was used to measure hardness and Young's modulus along the interfaces of Be/Cu bonded structure, The MPM was able to distinguish different fabrication techniques by a direct measurement of the hardness, Young's modulus, and H/E^2 which reflects the ability of deformation of the interfacial region.

A fundamental solution of plane elasticity in a finite domain is developed in this paper. A closed-form Green's function for the elastic field of an edge dislocation of arbitrary Burger's vector at an arbitrary point in an orthotropic finite electric domain, that is free of traction, is presented. the method is based on the classical theory of potential fields, with an additional distribution of surface dislocations to satisfy the free traction boundary conditions. A solution is first developed for a dislocation in a semi-infinite half-plane. The resulting field is composed of two parts: a singular contribution from the original dislocation, and a regular component associated with the surface distribution. The Schwarz-Christoffel transformation is then utilized to map the field quantities to a finite, polygonal domain. A closed form solution containing Jacobi elliptic functions is developed for rectangular domains, and applications of the method to problems of fracture and plasticity are emphasized

The Molecular Dynamics (MD) method is used to calculate defect energetics in ß-silicon carbide. Many-body interaction effects in this covalent material are accounted for by using a hybrid of two-body and three-body potentials. Calculated bulk properties of ß-SiC based on Uiis potential are in agreement with experimental data to within 17%. 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. The potential energy for the unperturbed computational cell is first calculated. The cell is Uien set at a defect configuration and relaxed, and the potential energy of the relaxed cell is calculated. Tlie difference between tlie potential energy of the unperturbed cell and that of the defect-containing cell is used to calculate the formation wid binding energies of point defects, defect clusters, and helium-vacancy clusters in SiC.

The lull spectrum of fracture toughness (J integrals), including pure mode I, different mixed mode I/III and pure mode 111, will be examined for a ferritic/martensitic stainless steel wilh 0.1C-8Cr-2W-0.2V-0.04Ta-Fe (by wt%), designated as F-82H. Tlie J integrals of pure mode I (J_{IC}) and mixed mode I/III (J_{mixed}) are determined with single specimen method using standard compact tension specimens and modified compact tension specimens, respectively. The pure mode 111 integral is measured with multiple specimen method using "triple-paillleg" specimens. Effects of hydrogen on the J integrals of pure mode I and mixed mode I/III are also going to be studied. 9 ppm H (about 500 appm) is pre-charged into specimens cathodically. The preliminary results showed Uiat addition of mode III Stress (shear stress) to mode I loading had a significant negative effect on the fracture toughness of F-82H.

The irradiation creep behavior of the fusion heats of HT9 and 9Cr-1Mo at –400°C has been measured to exposures as large as 208 dpa. HT9 is somewhat nonlinear in its response to hoop stress level in the range 0-200 MPa, but 9Cr-1Mo exhibits only slightly greater than linear behavior with stress level. The strain data of both alloys appear to include some contributions from precipitate-relaled density changes. Swelling may have occurred in 9Cr-1Mo.

Tensile and Charpy impact tests were made on three normalized-and-tempered 2 1/4Cr-2WV (0.1% C) steels with 0.1, *0.25*. and 0.5% V (all concentrations are in weight percent). Increasing vanadium from 0.1 to 0.25% increased the yield stress up to twenty percent. A higher ductile-brittle transition temperature (DBTT) accompanied the higher strength of the 0.25% V steel when both were tempered at 700°C. Tempering at 750°C gave similar DBTTs. Increasing vanadium from 0.25 to 0.5% caused a slight increase in strength with a large decrease in toughness. Thus a balance between strength and impact toughness is achieve with an intermediate vanadium concentration. Addition of 0.02% Ti to 2 1/4Cr-0.25V, 2 1/4Cr-2W, and 2 1/4Cr-2W-0.25V (0.1% C) steels caused a yield stress decrease of 10 to 30%, which was attributed to the effect of titanium on the MC precipitate distribution. The strength loss was accompanied by an increase in impact toughness, which may also have been affected by a decrease in prior austenite grain size. Furthermore, there was little difference in the DBTT of the Ti-modified steels tempered at 700 or 750°C. If it were possible to use a Ti-modified steel temperated at 700°C, this might offset the strength advantage of steels wilhout titanium, which have to be tempered at the higher temperature.

Experiments have shown that Type 316 stainless steel is susceptible to heat-affected-zone (HAZ) cracking upon cooling when welded using the gas tungsten arc (GTA) process under lateral constraint. The cracking has been hypothesized to he caused by stress-assisted helium bubble growth and rupture at grain boundaries. This study utilized an experimental welding setup, which enabled different compressive stresses to be applied to the plates during welding. Autogeneous GTA welds were produced in Type 316 stainless steel doped with 256 appm helium. The application of a compressive stress. 55 MPa, during welding suppressed the previously observed catastrophic cracking. Detailed examinations conducted after welding showed a dramatic change in helium bubble morphology. Grain boundary bubble growth along directions parallel to the weld was suppressed. The results suggest that stress-modified welding techniques may be used to suppress or eliminate helium-induced cracking during joining of irradiated materials.

At 178 dpa and ~400°C, the irradiation creep behavior of 20% cold-worked PCA has become dominated by the creep disappearance phenomenon. The total diametral deformation rate has reached the limiting value of 0.33%/dpa at the three highest stress levels. The stress-enhancement of swelling tends to camouflage the onset of creep disappearance, however.

Pure molybdenum and Mo-41wt% Re, in both the 20% cold-worked and aged and the annealed and aged conditions, were irradiated in FFTF/Mota to exposures as high as 111 dpa. Pure molybdenum appears to approach a saturation swelling level that is independent of the starting state. Cold-worked and aged molybdenum initially swells at a higher rate than that of solution-annealed and aged molybdenum and overshoots the saturation level at lower irradiation temperatures. This requires that part of the accumulated swelling be removed to approach saturation. probably by void shrinkage. The alloy Mo-41Re exhibits a more complex behavior with the annealed and aged condition initially swelling faster, but eventually the density change of both conditions begins to turn downward and tends toward densification. The role of solid transmutation to Tc, Re, and Os is thought to he very important in the irradiation behavior of these two metals. Calculations of transmutant generation are provided for FFTF, HFIR and STARFIRE spectra.

Irradiation has been completed in MOTA-2B lor creep tubes constructed from a candidate reduced activation austenitic alloy supplied by the Baikov Institute in Moscow. The total swains are strongly dependent on irradiation temperalure in tlic range 425-600°C, but are not completely linear with stress level.

Density change measurements have been complctcd on the Ispra second generation AMCR alloys irradiated in MOTA-2A. The various compositional modifications induce a wide **variation** of swelling behavior in the range 423-600°C at 32-36 dpa.

Previous **work** on a series of experimental high-manganese reduced-activation austenitic stainless steels demonstrated that they had improved tensile properties relative to type 316 stainless steel in both the annealed and 20% cold-worked conditions. Steels were tested with an Fe-20Mn-12Cr-0.25C (in weight percent) base composition, to which various combinations of Ti, W, V, P, and B were added. Tensile tests have now been completed on these steels after thermal aging at 600°C. Thermal suability varied with composition, but the alloys were as stable or more stable than type 316 stainless steel. The strength of *the* annealed stcels increased slightly after aging to 5000 h, while a strength decrease occurred **for** the cold worked steel. In both conditions, a steel containing a combination of **all** the alloying elements was most stable **and had** the best strength after thermal aging 5000 h at 600°C. Despite having much higher strength than 316 stainless steel after aging, the ductility of the strongest experimental alloy was **still** as good as that of316 stainless steel.

The effects of He/dpa ratio on swelling behavior were examined on three austenitic stainless steels. Materials were solution-annealed JPCA and two low carbon containing alloys (**C and K**) modified wilh titanium and niobium. These steels were neutron-irradiated in ORR and HFIR wilh and without spectrally tailoring, respectively. Achieved damage level was 7.4 **dpa** in ORR irradiation with average He/dpa of about 21 appm/dpa. In case of HFIR irradiation, Uiey were **33** dpa and 76 appm/dpa, respectively. Alloy to alloy variation and temperature dependence of swelling behavior are far more distinctively detected in ORR irradiation than in HFIR irradiation, in spite of tlie lower damage level of ORR irradiation. In the case **of** ORR-irradiation. **JPCA** exhibited small swelling values of <0.01 **and** 0.03% at ***603*** and 673 K. respectively, while a low carbon alloy K showed relatively larger swelling: 0.2% at 603 K and 0.6% at 673 K. Number densities of cavities in HFIR-irradiated alloys were larger than **those** observed in ORR by one to two orders. On the other band, number densities **and** sizes of dislocation loops produced by ORR irradiation were two to live tunes **as** large as those by HFIR irradiation. These facts suggest that in ORR condition with closer He/dpa to that of fusion, mutual annihilation rate of point defects was reduced and then bias driven cavity growth might be enhanced compared with HFIR condition.

The behavior of helium in welded joint fabricated using tungsten inert gas (TIG) welding process for a type 316 stainless steel wrapper tube irradiated in a fast reactor was investigated. The wrapper tube was irradiated to (1.5 - 4.2) x 10^{26} n/m^2 (helium level of 3 to 9 appm) at 395 - 410°C. All welded joints fractured in the heat-affected zone (HAZ). The microstructures of each portion of the base metal, the HAZ and the fusion zone in a welded joint were examined through a transmission electron microscope. Small helium bubbles were observed in number density of 2 x 10^{20} m^{-3} in the matrix and rarely found on the grain boundaries of the base metal. In the HAZ, small and large helium bubbles mixed and lined up along the grain boundaries. In particular, some of them elongated along the grain boundary. In the matrix of the fusion zone, delta-ferrite phases and unresolved carbides were scattered. Large cavities were attached to these precipitates and also occurred along grain boundaries. These results suggest that the failure in the HAZ of welded joints is attributed to the preferential growth and coalescence of helium bubbles in the grain boundaries of the HAZ caused by weld heat input and stress during welding

Microstructural data on the evolution of the dislocation loop, cavity, and precipitate populations in neutron-irradiated austenitic stainless steels are reviewed in order to estimate the displacement damage levels needed to achieve the "steady state" condition. The microstructural data can be conveniently divided into two temperature regimes. In the low temperature regime (below about 300°C) the microstructure of austenitic stainless steels is dominated by "black spot" defect clusters and faulted interstitial dislocation loops. The dose needed to approach saturation of the loop and defect cluster densities is generally on the order of 1 displacement per atom (dpa) in this regime. In the high temperature regime (~300 to 700°C). cavities, precipitates, loops in excess of 10 dpa are generally required to approach a "steady state" microstructural conditions. Due to complex interactions between the various microstructural components that form during irradiation, a secondary transient regime is typically observed in temperatures. This slowly evolving secondary transient may extend to damage levels in excess of 50 dpa in typical 300-series stainless steels, and to >100 dpa in radiation-resistant developmental steels. The detailed evolution of any given microstructural component in the high-temperature regime is sensitive to slight variations in numerous experimental variables, including heat-to-heat composition changes and neutron spectrum.

Candidate materials for first wall/blanket structures in ITER have been irradiated to damage levels of about 3 dpa at temperatures of either 60 or 250°C. Preliminary results have been obtained for several of these materials irradiated at 60°C. The results show that irradiation at this temperature reduces the fracture toughness of austenitic stainless steels, but the toughness remains quite high. The unloading compliance technique developed for the subsize disk compact specimens works quite well, particularly for materials with lower toughness. Specimens of materials with very high toughness deform excessively. and this results in experimental difficulties.

Dynamically compacted tungsten widi a starting density of 95.3% of the theoretical value densified 2 to 3% when irradiated in FFTF/MOTA-2A at three temperatures between 423 and 600°C and displacement levels corresponding to 32 aiid 36 dfpa in slainless steel. Rapidly solidified and dynamically compacted PCA with high levels of titanium and carbon were also irradiated at tliesc conditions. The density changes were sinall enough to determine that significant swelling had not occurred hut, microscopy is necessary to determine whether void growth occurred in addition to precipitate-related strains.

Nb-1Zr has been proposed lor potential application to ITER. Whereas previous irradiation studies on Nh-1Zr were focused on the annealed condition, this study involved a comparative irradiation of both the annealed and aged, and the cold-worked and aged conditions. Based on measurements of density change, the cold-worked aiid aged condition appears to first undergo a phase-related dilation prior to the onset of void swelling. while the annealed condition densified prior to swelling and in some cases docs not swell at all.

The corrosion and embrittlement of pure Nh, 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 tlie Nh-SV-I .25Zr alloy ("O" lot) exhibited both a modest corrosion rate and good resistance to embrittlement relative to other Nb-base alloys. At present, Nb-5V-1.25Zr is the most promising Nb-base alloy on the basis of both corrosion and embrittlement characteristics in HP deoxygenated water at 300°C.

Microstructural evolution associated with transmutation of ^{10}B lo helium and lithium has been characterized lo provide a better understanding of the boron-doping technique, frequently used to simulate the effect of helium generation under fusion reactor conditions. Transmission electron microscopy (TEM) was used lo examine specimens of V-20Ti alloy after irradiation at 600°C to ≈44-80 dpa in the Fast Flux Test Facility (FFTF). In the earlier stage of irradiation to low fluence, concentric shells of He-darnage and Li-damage zones are produced around a V_3B_2 precipitate or a ^{10}B-rich cluster. On further irradiation, helium atoms diffuse away froin the damage shell either to be dissolved in the matrix or to form microcavities, leaving a shell rich in Li, defect clusters, and dislocations. Oxygen atoms in solid solution migrate toward the Li-rich shells, and γ-LiV_2O_5 shells precipitate subsequently. In view of this behavior. neither boron nor Li produced froin tlic transmutation is likely to result in a detrimental weakening of grain boundaries.

Irradiation of the seven DHCE capsules was completed in the Materials Open Test Assembly (MOTA)-2B at the end of Cycle 12B in the Fast Flux 'lest Facility (FFTF). The accrued exposure was 203.3 effective lull-power days (EFPDs), vis-a-vis the target exposure **of** 300 EFPDs. Peak damage in the samples was ≈29 displacement per aloin (dpa). All seven capsules have been discharged from the FFTF and are being shipped to Argonne National Laboratory (ANL), where the samples will be retrieved lroin the capsules and distributed to the experimenters. including Monbusho of Japan, for examination and testing. **A** substantial effort **is** underway at ANL to retrieve the samples from the highly tritiated capsules.

The thermal creep rates and stress-rupture life of V-5Cr-5Ti and V-10Cr-5Ti alloys were determined at 600°C and tlic impurity composition and microstructural characteristics of creep-tested *specimens* were *analyzed* and correlated with the measured creep properties. The results of these tests show that V-5Cr-5Ti, which contains impurity compositions typical of a commercial vanadium-base alloy, exhihits creep strength substantially supcrior to that of V-20Ti, HT-9, or Type 316 stainless steel. The V-10Cr-5Ti alloy exhibits creep strength somewhat higher than that of V-5Cr-5Ti.

Ductile-brittle transition temperatures (DBTTs) wcrc determined hy Charpy-impact tests for dehydrogenated and hydrogenated V-3Ti, V-5Cr-3Ti, and V-5Cr-5Ti alloys. These DBIT data complement the data previously obtained hy Loomis et al. on Charpy-impact testing of unalloyed V, V-1Ti, V-3Ti-1Si, V-5Ti, V-10Ti, V-18Ti, V-4Cr-4Ti, V-8Cr-6Ti, V-9Cr-5Ti, V-10Cr-9Ti, V-14Cr-5Ti, V-15Cr-5Ti, V-7Cr-15Ti, and Vanstar-7 alloys. The results show that V alloys with Ti additions (0-18 wt.%) have a minimum DBTT (≈-250°C) in an alloy containing 3-5 wt.% Ti, that addition of 4 to 15 wt.% Cr to V-(4-6)Ti alloy results in a substantial increase (25-215°C) of the DBTT, and that *0.5* and 1.0wt.% Si additions to V-3Ti alloy result in a significant increase (−100°C) in DBTT. In addition, the rcsults show that the presence of 400-1200 appm H in unalloyed V and V-base alloys causes a significant increase (≈400 ppin 0, ≈200 ppin C, and ≈900 ppm Si for use as structural material in a fusion reactor.

Irradiation continues in the EBR-II and HFIR reactors of pure copper and GlidCop CuAl25 in the form of subsize tensile fatigue specimens. The **list** phase of the EBR-II irradiation sequence has been completed. **A** size effects experiment conducted on unirradiated CuAl25 fatigue specimens is nearing completion. Early results on the fatigue behavior of **suhsize** spcciinens are presented in this report.

Various oxide-dispersion-strengthened copper alloys have been irradiated to 150 dpa at 415°C in the Fast Flux Test Facility (FFTF). The Al_2O_3-strengthened GlideCop^{TM} alloys, followed closely by a HfO_2-strengthened alloy, displayed the best swelling resistance, electrical conductivity, and tensile properties. The conductivily of the HfO_2-strengthened alloy reached a plateau at die higher lcvels of irradiation, instead of exhibiting the steady decrease in conductivity observed in the other alloys. A high initial oxygen content resulted in significantly higher swelling for a series of castable oxide-dispersion-strengthened alloys, while a Cr_2O_3-strengthened alloy showed poor resistance to radiation.

Density measurements have been completed on copper alloys irradiated in MOTA 2A and MOTA 2B at (375°C, 12.7 dpa wid 21.2 dpa) and (423°C, 48.0 and 95.4 dpa). While most of the density changes ohserved are consistent with those of earlier studies, there were several surprises. Addition of 5% Ni appears to accelerate the swelling rate initially at 423°C, but depresses swelling at 375°C. The suppressing action of cold work on swclling of Cu-5Ni is relatively small, and Cu-5Mn resists swelling very strongly in both llie annealed and cold-worked conditions.

A total of 74 allays of varying composition and thermomechanical condition have been prepared for a joint US-Russia irradiation experiment in llic SM-3 reactor in Dimitrovgrad, Russia. The alloys will be irradiated in the fonn of TEM disks and sheet tensile specimens at temperatures of about 120, 250, and 340°C lor one 45-day cycle in the core and Channel 2 irradiation positions. This will produce damage lcvcls of about 7 and 1 dpa, respectively. Cadmium shielding will he used in the Channel 2 position to shield the thennal neutrons and thereby reduce the amount of solid transmutation products in copper.

Transmission electron microscopy was used to compare the microstructure and particle distributions of two commercial oxide dispersion-strengthened copper alloys, GlidCop A125 and MAGT 0.2. Measurements were made on specimens in their as-wrought condition, after thermal annealing for 1 h at 900°C, and after 3 MeV Ar^+ ion irradiation at 180 and 350°C to damage levels of 20 to 30 displacements per atom (DPA). All of the annealed and ion-irradiated specimens were found to be resistant to recrystallization. In addition, void formation was not observed in any of the irradiated specimens. The GlidCop oxide particle geometry was transfonned from triangular platelets to circular disks by the ion irradiation. The MAGT parlicle geometry consisted of circular disks and spheres before and after inadiation. The oxide particle edge length int he unirradiated GlidCop alloy was abut 10 nm, whereas the mean particle diameter in both the MAGT and GlidCop alloys was about 6 nm.

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The plugged pipe removed from the cold-trap purification loop of the forced-circulation lithium system was examined to study mass transfer/deposition behavior and to establish the cause of plugging. Several intermetallic compounds were identified in residue collected from the plugged cold-trap pipe. Plugging was caused by deposition of calcium/zinc/nickel crystals in the pipe sections within the pump coil and flow-meter magnet. Addition of calcium as a getter to reduce the conccnuation of nitrogen in the lithium leads to formation of $(Ca,Zn)Ni_5$ crystals and subsequent plugging of the cold-trap loop. Deposits of manganese/iron/nickel globules and manganese/zinc/nickel dendrites, as well as Li_9CrN_5 and possibly Ca_3N_2, were also identified in the residue. These phases may have reduced flow through the cold-trap loop hut were not abundant enough to plug the loop. The results indicate that the use of a dissolved getter, such as calcium, to reduce nitrogen content in an austenitic stainless steel loop may not he effective. Elements in the lithium from structural alloys (e.g., due to corrosion) aiid those added (e.g., calcium) to reduce the concentration of nonmetallic impurities (e.g., nitrogen) play an important role in the mass transfer/deposition behavior in circulating lithium systems.

Scoping tests were conducted on compatibility of gallium with candidate structural materials. e.g., Type 316 SS, Inconel 625, aid Nb-5 Mo-1 Zr alloy, as well as Armco iron, Nickel 270, and pure chromium. Type 316 stainless steel is least resistant and Nh-5 Mo-l Zr alloy is most resistant to corrosion in static gallium. At 400°C, corrosion rates are ≈4.0, 0.5, and 0.03 mm/y for Type 316 SS, Inconel *625*, and Nb-5 Mo-1 Zr alloy, respectively. The pure metals react rapidly wilh gallium. In contrast to findings in earlier studies, pure iron shows greater corrosion Uian does nickel. 'llie corrosion rates at 400°C are ≥90 and 17 mm/y, respectively, for Armco iron and Nickel 270. 'The results indicate Uiat at temperatures up to 400°C, corrosion occurs primarily by dissolution accompanied by formation of metal/gallium intermetallic compounds.

Susceptibility of crevice-weldincnt specimens of Types 316L and 316NG stainless steel (SS) to SCC was investigated in slow-strain-rate-tensile (SSRT) tests in watcr that simulates important parameters anticipated in first-wall/blanket systems. The SSRT tests were performed in oxygenated water containing 0.06-10 pprn chloride at temperatures of 95 to 225°C to estahlish the effects of water purity and temperature on SCC resistance. These steels, including weldments, exhihit good resistance to SCC under crevice conditions at temperatures of <150°C in water containing ≤0.1 ppin chloride. It appears that Type 316NG SS is somewhat more resistant to SCC than Type 3161, SS at temperatures >150°C in oxygenated water containing 0.1-10 ppin chloride. Most specimens fractured in the base metal, and several others fractured in the heat-affected zone (HAZ) of the weld, but non failed in the weld metal.

Aluminide coatings were produced on vanadium and vanadium-base alloys by exposure of the materials to liquid lithium that contained 3-5 at.% dissolved aluminum in sealed capsules at temperatures between 775 and 880°C. Reaction of the aluminide layer with dissolved nitrogen in liquid lithium provides a means of developing an in-situ electrical insulator coating on the surface of the alloys. The electrical resistivity of AIN coatings on aluminided V and V-20 wt.% Ti was determined in-situ.

Crack-growth-rate (CGR) tests were performed on 1-in.-thick (1T) compact-tension (CT) specimens of Types 316NG and 316L stainless steel (SS) in oxygenated water conraining 0-5 ppin Cl^- at 150, 185, and 225°C. The results obtained under cyclic loading conditions at stress intensity factors of ≈27 to 39 $MPa \cdot m^{1/2}$ indicate that environmental enhancement of the rates increases with Cl^- concentrations >0.1 ppm at 150°C in comparison with calculated rates in air under the specific loading conditions. In contrast, at the higher temperatures the CGRs were not affected by Cl^- in oxygenated water but were greater than the predicted rates in air by one order of magnitude.

Aluminide coatings were produced on vanadium and vanadium-base alloys by exposure of the materials to liquid lithium that contained 3-5 at.% dissolved aluminum in sealed V and V-20 wt.% Ti capsules at temperatures between 775 atid 880°C. After each test, the capsules werc opened and the samples were examined by optical microscopy and scanning electron microscopy (SEM), and analyzed by electron-energy-dispersive spectroscopy (EDS) and X-ray diffraction. Hardness of the coating layers and hulk alloys was determined by microindentation techniques. The nature of the coatings, i.e., surface coverage, thickness, and composition, varied with exposure time and temperature, solute concentration in lithium, and alloy composition. Solute elements that yielded adherent coatings on various substrates can provide a means of developing in-situ electrical insulator coatings by reaction of the reactive layers with dissolved nitrogen in liquid lithium.

No contributions

The key radiation effects issues associated with the successful operation of ceramic materials in components of the planned International Thermonuclear Experimental Reactor (ITER) are discussed. Radiation-induced volume changes and degradation of the mechanical properties should not be a serious issue for the fluences planned for ITER. On the other hand. radiation-induced electrical degradation effects may severely limit allowable the exposure of ceramic insulators. Degradation of the loss tangent and thennal conductivity may also restrict the location of some components such as ICRH feedthrough insulators to positions far away from the fust wall. In-situ measurements suggest Uiat the degradation of physical propenies in ceramics during irradiation is greater than Uiat measured in postirradiation tests. Additional in-situ data during neutron irradiation are needed before engineering designs for ITER can he finalized.

Cross section transmission electron microscopy was utilized to examine the ndiation-induced microstructural changes in oxide ceramics after in-adintion with a wide variety **of** ion beams. The microstructure showed **a** strong dependence on ion mass and energy. The microstructural results have been correlated with the calculated depth-dependent partitioning between ionization and displacement damage. This correlation indicates that defect clusters do not form **in** $MgAl_2O_4$ if the ratio of encrgy deposited into electronic ionization to atomic displacements is greater than about 10. The corresponding ratio needed to suppress defect cluster fonnation in MgO and Al_2O_3 is 500 to 1000. Additional microstructural evidence obtained on the ion irradiated ceramic specimens suggests that the physical mechanism responsible for the **lack** of defect clusters in highly **ionizing** radiation environments is associated with ionization-enhanced diffusion (IED), which promotes annihilation of the point defects created **by** displacement damage **during** the irradialion. The **most** important parameter for IED is the ratio of ionizing to displacive radiation, since Uiis is roughly proportional to **Ute** amount of ionization per dpa. Ilowever, the absolute magnitude of the ionizing radiation **flux** is also important.

The initial results are presented from a cyclotron ion irradiation program investigating radialion-induced conductivity (RIC) and radiation-induced electrical degradation (RIED) of ceramic insulators. Polycrystalline specimens of Al_2O_3, $MgAl_2O_4$, AIN and Si_3N_4 were irradiated with eilher 28 MeV He^{++} or 20 MeV H^+ ions at temperatures between 150 **and** 600°C with an applied dc electric field ***of*** 100 **to** 500 V/mm. **A** large prompt increase in the electrical conductivity was observed in all of the specimens during irradiation. However, there was no evidence for permanent electrical degradation **in any** of the specimens for damage levels up to **about 5** x 10^{-3} displacements **per** atom.

$MgAl_2O_4$ **spinel** specimens irradiated in FFTF-MOTA **at** temperatures between 385 and 750°C to fluences ranging from ***2.2*** to **24.0** x 10^{22} n cm^{-2} (E>0.1 MeV) darken significantly, hut do not develop any **loss** in weight or change in dimensions. Measurements of **Knoop hardness** and **its** dependence on crystalline orientation, neutron fluence and irradiation temperature are in progress. Measurements of elastic properties are also nearing completion.

Si/SiC composites exhibit novel mechanical properties relative to their monolithic counterparts. The crack velocity (da/dt) versus **stress** intensity (K) relationship for monolithic ceramics can be described by a simple power law relationship where SiC/SiC was found to exhibit a multi-stage da-dt versus K relationship similar to that for stress corrosion of metals. A K independent stage II was followed by a strongly K dependent state III, which paralleled the monolithic behavior. Experiments to determine the threshold K or stage I were not conducted; however, it is expected that they exist for these materials. There is also evidence that the fracture resistance of these materials is greater if cracks are produced by subcritical growth processes relative to machined notches. Oxygen was found to increase da/dt and decrease the K for the stage II to stage III transition while cyclic loads produced little damage at low K values but there was some evidence for increasing damage with increasing number of cycles and K.

An irradiation experiment was carried out during the summer of 1992 at the Los Alamos Spallation Radiation Effects Facility (LASREF). *In situ* measurements of electrical conductivity in alumina, sapphire, and mineral-insulated electrical cables were made at 640°C, 590°C, and 400°C. Both DC and AC (100 Hz to 1 MHz) measurements were made to a fluence of approximately 3×10^{23} n/m^2. Optical absorption from 300 nm to 800 nm was measured in pure silica- and OH-doped silica-core optical fibers during the irradiation. A large number of passive samples were included in the irradiation, some at the furnace temperatures and some at ambient temperature. This report describes preliminary analysis of the DC conductivity measurements. The AC measurements are analyzed in the companion report.

Samples of single crystal Al_2O_3, commonly known as sapphire, and polycrystalline Al_2O_3 were irradiated with spallation neutrons at the Los Alamos Spallation Radiation Effects Facility (LASREF) under various temperature conditions and with a continuously applied alternating electric field. This paper describes the results of measurements on the sapphire samples. Neutron fluence and flux values are estimated values pending recovery and analysis of dosimetry packages. The conductivity increased approximately with the square root of the neutron flux at fluences less than 3×10^{19} n/m^2 at fluxes less than 1×10^{16} n/m^2-sec. Conductivity initially decreased at low fluences with minimums near fluences of 1×10^{20} n/m^2. Incubation periods with a gradual increase in conductivity preceded the onset of an accelerated increase in conductivity beginning at fluences as low as 10^{21} n/m^2. The increase in conductivity reached saturation levels as high as 2×10^{-2} (ohm-m)$^{-1}$ at fluences as low as 2×10^{22} n/m^2. Frequency swept impedance measurements indicated a change in the electrical properties from capacitive to resistive behavior with increasing fluence.

Ceramics subjected to irradiation show a wide variety of damage responses, depending on composition, nature of bonding, crystal structure, impurity levels, starting microstructure, number of phases, and type of bombarding particle. As doses reach high levels (a condition that varies in magnitude from one material to another) major changes in physical properties can occur, and atomic arrangements may even change from crystalline to disordered. However, some ceramics show marked resistance to damage, and some properties may improve. More work is needed to fully understand these phenomena, but it is currently possible in many cases to predict at least qualitatively both microstructural damage response and observed property changes.

Microwave heating of plasmas in fusion reactors requires the development of microwave 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 papers have addressed this question at higher temperatures and at low temperatures. The current paper extends the low temperature calculations to determine the effect of phonon-electron scattering on the thermal conductivity at 77 K due to RIC. 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.

Surface preparation has an observable effect on the data obtained from die Nanoindenter for shallow (20 nm) indents on CVD SiC when polished with Syton™. This observed effect is significantly less for 1/2 micron diamond polished CVD SiC and for **deep** (160 nm) indents. These effects were manifested by tlic relative variations in the experimental modulus and hardness data. **An** analytical analysis of the anticipated variation in the modulus and hardness is performed and shown to correlate well with the observed trends. The observed variations appear to he the result of SiC material properties, as **well** as surface preparation, since the predicted variations are much smaller than the ohserved variations.

A micro mechanical model is developed to **study** fiber pull-out and crack bridging in fiber reinforced Sic-Sic composites with time dependent thermal creep. By analyzing the **creep data** for monolithic CVD SiC (matrix) **and** the SCS-6 SiC fibers in the temperature range 900-1250°C, it is found that the matrix creep rates can hc ignored in comparison to those ***of*** fibers. Two important relationships are obtained: (1) a lune dependent relation hetween the **pull-out** stress and the relative sliding distance between the fiber and matrix for the purpose of analyzing pull-out experiments, arid **(2)** the relations between the bridging **stress** and the **crack** opening displacement to he **used** in studying the mechanics and stability of mauix crack bridged by fihcrs at high temperatures. The present analysis can be also applied **to** Nicalon-reinforced CVD SiC matrix system since the Nicalon fibers exhibit **creep characteristics** similar to those of the SCS-6 fibers.