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New data on radiation-induced hardening, low-temperature creep and potential susceptibility (sensitization) to aqueous corrosion have been obtained on various heats of austenitic stainless steel (including type 316) irradiated at 60 to 400°C to 7 to 13 dpa. The data were obtained from spectrally tailored reactor experiments, whose radiation damage parameters are similar to those in the proposed International Thermonuclear Experimental Reactor (ITER) first-wall (FW) and blanket design. Austenitic stainless steels were found to increase significantly in strength at 60 to 330°C, to have higher irradiation-creep rates at 60°C than at 200 to 400°C, and to show radiation-induced changes in electrochemical properties at 200 to 400°C. These data on several radiation-induced property changes suggest that type 316 steel may be an adequate material for the FW of ITER. However, there is a definitely a need for new data on fracture-toughness and on fatigue behavior below 400°C, as well as more data on irradiation-creep and effects of irradiation on corrosion properties, to better define temperature and dose dependencies for more detailed design analyses. Cold-working should remain an optional as-fabricated condition for the FW of ITER. Many properties of SA and CW 316 become similar after irradiation at 60 to 400°C. The higher initial yield-strength of CW 316 will allow higher design stress and elastic strain limits.

6.2.2	SWELLING AND MICROSTRUCTURAL ANALYSIS OF U.S.-PCA AUSTENITIC STAINLESS STEEL IRRADIATED AT 60 TO 400°C IN ORR SPECTRAL-TAILORED EXPERIMENTS -- (Oak Ridge National Laboratory) .....	99
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6.2.3	STATUS OF MOTA IRRADIATION EXPERIMENTS ON REDUCED ACTIVATION AUSTENITIC ALLOYS (Pacific Northwest Laboratory, Westinghouse Hanford Company, Oak Ridge National Laboratory, Hokkaido University, Nagoya University, Ispra Establishment, and Baikov Institute) .....	109
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A number of collaborative international experiments are being conducted to assess the feasibility of Fe-Cr-Mn austenitic steels for fusion service. A review is presented of the current status of these various efforts. The first time irradiation of an Fe-Cr-Mn steel in the form of pressurized tubes is highlighted.

6.2.4	PHASE STABILITY IN THERMALLY AGED Fe-Cr-Mn ALLOYS (Northwest College and University Association for Science, Washington State University, and Pacific Northwest Laboratory) .....	113
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Fe-Cr-Mn alloys have been proposed as a structural material for fusion reactors that will exhibit reduced long-term radioactivity. However, Fe-Cr-Mn ternary equilibrium has not previously been well defined at temperatures below 650°C. The current experiment characterized phase evolution and equilibrium in three alloys; Fe-15Cr-15Mn, Fe-10Cr-30Mn and Fe-30Mn in the 20% cold-worked condition aged from 300 to 700°C for 1,000 to 30,000 hours at temperatures from 300 to 600°C. Results indicate that phase formation is extremely sluggish, and cold-working and long aging times are required to initiate  $\sigma$  phase formation in alloys not previously predicted to exhibit such precipitation.  $\sigma$  phase appears to form on high diffusivity paths such as recrystallizing grain fronts and grain boundary triple points. Three isothermal sections at 300, 500 and 600°C were constructed from the experimental data that extend the temperature range of the Fe-Cr-Mn ternary diagram to lower temperatures than previously published.

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Microstructural evolutions in V, V-Cr, V-Ti, V-Ti-Cr, and V-Ti-Si alloys have been characterized by TEM after irradiation at 420°C in FFTF/MOTA. The results have been correlated with swelling behavior of the alloys to provide a better understanding of the superior resistance of some of the high-Ti heats to void swelling.

**6.3.2 TENSILE PROPERTIES OF VANADIUM AND VANADIUM-BASE ALLOYS (Argonne National Laboratory) . . . . . 145**

Tensile property data are presented for unalloyed vanadium and **18** vanadium alloys that are considered candidates for structural material in a magnetic fusion reactor. The compositions of these candidate alloys are principally those vanadium binary and ternary alloys with Cr, Ti, and Si additions. The dependence of the tensile properties for these materials on temperature (**25-700°C**), Cr concentration (0-15%), Ti concentration (0-20%), and Si concentration (0.05-1.28%) is presented in the form of tables, graphs, and parametric equations. The strengthening of V by Cr, Ti, and Si additions is discussed with emphasis on the role of interaction of these alloying additions with interstitial impurities (i.e., O, N, and C).

**6.3.3 REVISED CALCULATIONS FOR THE DYNAMIC HELIUM CHARGING EXPERIMENT IN FFFTF/MOTA 2B (Pacific Northwest Laboratory) . . . . . 156**

Calculations of the initial tritium loading, lithium isotopic ratio, and helium pressure have been revised for the planned dynamic helium charging experiment (**DHCE**) in FFFTF/MOTA-2B based on current neutronic information regarding the neutron flux spectrum.

**6.3.4 STATUS OF THE DYNAMIC HELIUM CHARGING EXPERIMENT (DHCE) IN FFFTF/MOTA (Argonne National Laboratory, Tohoku University, Pacific Northwest Laboratory, and Westinghouse Hanford Company) . . . . . 159**

Preparation of the **DHCE** experiment has been completed and it has been inserted into MOTA 2B for irradiation in FFFTF. The experiment will produce helium in test specimens of vanadium alloys to simulate a fusion environment.

**6.4 COPPER ALLOYS . . . . . 165**

**6.4.1 MICROSTRUCTURAL EXAMINATION OF PURE COPPER AND CUJNi IRRADIATED WITH FISSION OR SPALLATION NEUTRONS (Pacific Northwest Laboratory, Riso National Laboratory, and Los Alamos National Laboratory) . . . . . 167**

Specimens of pure copper and a copper alloy containing 5 atomic percent nickel have been examined by transmission electron microscopy following irradiation by neutrons in a mixed thermal reactor, a fast reactor and a spallation neutron source. Radiation damage microstructures have been investigated in order to identify microstructural differences and to determine suitability of copper alloys for high heat flux substructure applications in a fusion device. Most conditions developed void swelling, and all were found to contain defect clusters on a fine scale. Damage accumulation in Cu-5Ni was found to be significantly different from that observed in pure copper. Based on limited comparisons of irradiation at 335°C, high energy spallation neutrons do not appear to have any significant effect on swelling at low dose. The results of this study confirm the need for improved temperature control of irradiation experiments in order to correlate radiation damage to fusion conditions.

**6.4.2 STATUS OF COPPER IRRADIATION EXPERIMENTS (Pacific Northwest Laboratory, University of Missouri-Rolla, Riso National Laboratory, University of Illinois, Tohoku University, Oak Ridge National Laboratory, and SCM Metal Products) . . . . . 186**

A variety of FFFTF irradiation studies and out-of-reactor studies are in progress to determine the response of copper alloys to the environment anticipated for fusion service. An updated review of these various studies is presented.

**6.4.3 NEUTRON-INDUCED CHANGES IN DENSITY OF COPPER ALLOYS (Pacific Northwest Laboratory and Westinghouse Hanford Company) . . . . . 192**

Density change measurements have been completed on the Generation 2.0 copper alloy experiment at 411°C after reaching 100 dpa. The Glidcop alloy CuAl25 continues to exhibit excellent resistance to void swelling. Welding and high oxygen levels both degrade the swelling resistance of oxide dispersion-strengthened alloys. The alloy Cu-2.0 Be also resists swelling and appears to be densifying in response to the continued formation of the transmutant nickel.

**6.4.4 RADIATION-INDUCED CHANGES IN ELECTRICAL CONDUCTIVITY OF A WIDE RANGE OF COPPER ALLOYS (Pacific Northwest Laboratory, University of Illinois, and Tohoku University) . . . . . 199**

A wide variety of radiation-induced changes in electrical conductivity was observed in a series of irradiation experiments conducted on copper alloys in FFFTF/MOTA. The behavior of each alloy was found to depend on the alloy composition, starting state, irradiation temperature, and the sometimes complex interaction of three radiation-driven processes. These processes are transmutation, void swelling, and solute redistribution.

## 6.4.5 BRAZING OF COPPER-LUMINA ALLOYS (Auburn University and Oak Ridge National Laboratory) . . . . . 206

An induction braze has been developed to join copper-alumina alloys to eliminate the requirement for either plating the joining surfaces prior to brazing or the use of an inert cover gas or a vacuum pump to prevent excessive oxidation. Preliminary tensile tests of induction lap joints and fatigue tests of induction brazed butt joints have been conducted. A theoretical study of the diffusion of silver into the alloy has been compared to experimental EDX analysis

## 7.0 ENVIRONMENTAL EFFECTS ON STRUCTURAL MATERIALS . . . . . 213

### 7.1 ENVIRONMENTAL EFFECTS ON SiC/SiC COMPOSITES FOR FUSION STRUCTURAL APPLICATIONS (Pacific Northwest Laboratory) . . . . . 215

The chemical stability of SiC/SiC composites in fusion relevant environments has been evaluated from the database available in the literature for monolithic SiC. The results of this assessment suggests that the primary chemical reactions that will limit the stability of these materials are: 1) Li reduction of surface and grain boundary glass phases in a liquid Li coolant, 2) impurity effect on oxidation and the oxidation of the fiber/matrix interfacial layer in a He coolant and 3) H<sub>2</sub> reduction of the "passive" SiO<sub>2</sub> layer and the reaction of the H<sub>2</sub> with SiC to form gaseous reaction products. Strength degradation has been observed for each of these reactions under certain conditions, but the database is insufficient to predict whether similar property changes will occur in fusion relevant environments. Clearly, more experimental data is needed to evaluate the stability of SiC/SiC composites for long-term exposures in fusion relevant environments.

### 7.2 EFFECT OF GAMMA IRRADIATION ON STRESS-CORROSION BEHAVIOR OF AUSTENITIC STAINLESS STEEL UNDER ITER-RELEVANT CONDITIONS (Pacific Northwest Laboratory) . . . . . 231

This study wupled with other published data supports the conclusion that gamma irradiation will not induce Stress corrosion cracking if the oxygen activity in the ITER water coolant is maintained at a low level. This conclusion is based on stress corrosion cracking tests conduned at 100°C in deionized water with 10 ppm Cl<sup>-</sup> on solution annealed and sensitized Type 316 SS and PCA. The material was sensitized to about 5 C/cm<sup>2</sup> and fatigue crack growth tests were conducted in an autoclave exposed to a <sup>60</sup>Co source with tests conduned at 0, 2.3 x 10<sup>2</sup> and 6.5 x 10<sup>3</sup> rad/h. The material and water chemistry conditions were chosen to represent the worst case condition expected in an ITER resulting from poor welding and water chemistry control. Crack growth rates were found to decrease by about a factor of 2 in the presence of both gamma fluxes. Average crack velocities were 2.0 and 1.5 x 10<sup>-5</sup> mm/cycle for Type 316 SS and PCA, respectively, in the absence of gamma irradiation and 1.3 and 0.74 x 10<sup>-5</sup> mm/cycle at both gamma fluxes. These results suggest, for the limited conditions examined in this study, that radiolysis alone will not cause stress corrosion cracking of slightly sensitized Type 316SS or PCA in water with 10ppm Cl<sup>-</sup> at 100°C. Other conditions that have not been examined and may cause crack growth include: radiolysis effects in high O<sub>2</sub> activity water, hydrogen induced cracking of radiation hardening Type 316 SS and PCA and irradiation-assisted Stress corrosion cracking (IASCC).

### 7.3 RADIATION-INDUCED SEGREGATION IN IRRADIATED TYPE 304 ALLOYS FOR THE ICG-IASCC ROUND ROBIN - (Oak Ridge National Laboratory) . . . . . 237

Grain boundary RIS in two neutron-irradiated type 304 stainless steels has been investigated by X-ray microanalysis. In the wventional alloy (LC), narrow (≤5 nm width) RIS zones depleted in chromium and iron and enriched in silicon, phosphorus, and nickel were observed near grain boundaries. For the higher purity alloy (QC), similar width (≤6 nm) RIS zones depleted in chromium and iron and enrichment in nickel were observed at boundaries, though at reduced magnitudes relative to those in the LC material. No significant segregation of siliwn or phosphorus was observed in the QC material. RIS zones associated with faulted dislocation loops in the LC material were detected and shown to be depleted in chromium and enriched in nickel and iron relative to the matrix.

### 7.4 AQUEOUS STRESS CORROSION OF CANDIDATE AUSTENITIC STEELS FOR ITER STRUCTURAL APPLICATIONS (Argonne National Laboratory) . . . . . 241

Hydrogen embrittlement of candidate first-wall materials (e.g., stainless steel, or SS) is a key issue for ITER because of hydrogen production due to (n,p) transmutation reactions; hydrogen generation or up to 2500 appm is anticipated. Material composition, changer in composition caused by transmutation reanions, irradiation damage, environment, and expected loading wnditions were reviewed to assess the relative importance of various degradation processes for a fusion reactor first wall. The stability of austenite with regard to creation of deformation- or stress-induced manensile, α', was considered in relation to SCC and hydrogen-assisted crack growth in SSs at low temperatures (<150°C).

## 7.5 PRELIMINARY ASSESSMENT OF AQUEOUS CORROSION OF NIOBIUM ALLOYS FOR STRUCTURAL APPLICATIONS IN THE ITER DIVERTOR (Argonne National Laboratory) . . . . . 248

Niobium and Nb-base alloys are under consideration as candidate materials for the **ITER** divertor structure. Unalloyed Nb and binary **alloys** of Nb-2.5Ti, Nb-2.5Zr, Nb-2.5Hf, Nb-2.5V, Nb-2.5Ta, Nb-2.4Mo, Nb-2.5W, Nb-2.5Fe, **Nb-2.5Ti**, and ternary **alloys** of Nb-2.5Ti-2.5Ta, Nb-2.5Ti-2.5Mo, Nb-2.5Hf-2.5Mo (expressed in terms of atomic percent alloy addition) **were exposed to** high-purity water with < 30 ppb dissolved oxygen in a refreshed Stainless steel autoclave (with a 1300-psi overpressure) to determine the extent of oxidation as a function of time at 300°C. Corrosion data are presented for pure Nb and for ten binary and three ternary Nb-base alloys exposed for up to 2906 h. All alloys exhibited weight gains except Nb-2.5Ni, **Nb-2.5Ta**, and Nb-2.5Fe, which lost weight after 0, 1500, and 2200 h, respectively. Of the alloys that gained weight, the V and Mo additions **were** the most effective in minimizing weight gain. The ternary alloys incorporating **Mo** were superior in terms of bath resistance to weight gain and embrittlement.

## 8.0 SOLID BREEDING MATERIALS . . . . . 257

### 8.1 IRRADIATION EXPERIMENT DESIGN FOR IN SITU TRITIUM RECOVERY FOR $\text{Li}_2\text{O}$ and $\text{Li}_2\text{ZrO}_3$ : BEATRIX-II, PHASE II (Pacific Northwest Laboratory, Japan Atomic Energy Research Institute, AECL Chalk River, and Westinghouse Hanford Company) . . . . . 259

BEATRIX-II is an irradiation experiment designed to study the in situ tritium release behavior from **selected** ceramic **solid** breeder materials. The second irradiation cycle of the experiment. Phase II, will include a temperature change capsule with a ring specimen of  $\text{Li}_2\text{O}$  and a temperature gradient capsule with  $\text{Li}_2\text{ZrO}_3$  spheres. The temperature change capsule is designed to achieve temperatures in the range from 485 to 650°C while the temperature gradient capsule will include a temperature range from 450 to 1200°C. The effect of specimen temperature, sweep **gas** composition, irradiation damage, and sweep gas **flow** rate on the tritium recovery behavior will be determined in a fast neutron flux to burnups of 8%.

### 8.2 IN SITU TRITIUM RECOVERY FROM $\text{Li}_2\text{O}$ IRRADIATED UNDER A LARGE TEMPERATURE GRADIENT BEATRIX-II SOLID SPECIMEN (Japan Atomic Energy Research Institute, Pacific Northwest Laboratory, and AECL Chalk River) . . . . . 270

BEATRIX-II is an in situ tritium recovery experiment to determine the tritium release characteristics of  $\text{Li}_2\text{O}$  in a fast neutron flux. A **large** diameter specimen of  $\text{Li}_2\text{O}$  has been irradiated under a steep temperature gradient to 4% burnup in a **variety** of sweep **gas** compositions and **flow** rates. Decreasing the amount of hydrogen in the sweep **gas** significantly decreases the tritium recovery rate. During irradiation the temperature of the specimen has remained stable and the results **suggest** that  $\text{Li}_2\text{O}$  is a viable fusion solid breeder material.

### 8.3 INVESTIGATION OF TRITIUM RELEASE AND RETENTION IN LITHIUM ALUMINATE - (Argonne National Laboratory and Commissariat a L'Energie Atomique) . . . . . 280

Tritium release from single crystal lithium aluminate has been investigated by a series of **isothermal** anneal and constant rate heating **experiments**. The results of these **experiments** indicate that after anneals at **low** temperature, a large fraction of the tritium present before the anneal remains in the sample. We **have** modeled this behavior based on first order release from three **types** of sites. At low temperature the release is dominated by one site. while the tritium in the other **sites** is retained in the solid. Dopants appear to alter the distribution of tritium between the sites. Adding MgO decreases the fraction of tritium released at 777°C, while increasing the fractions released at 538 and 950°C.

### 8.4 DESORPTION CHARACTERISTICS OF THE $\text{LiAlO}_2\text{-H}_2\text{-H}_2\text{O(g)}$ SYSTEM - (Argonne National Laboratory) . 284

The energetics and kinetics of the evolution of  $\text{H}_2\text{O}$  and  $\text{H}_2$  from  $\text{LiAlO}_2$  have been studied by the temperature programmed desorption technique. After treating the sample with helium containing 990, 495, or 247 vppm  $\text{H}_2$  at 923 K,  $\text{H}_2\text{O}$  and  $\text{H}_2$  evolution was observed during 473 to 1073 K (200 to 800°C) ramps at rates of 5.6 K/min. The sweep **gas** was either pure helium or the same He-H<sub>2</sub> mixture that **was** used in the preliminary treatment. The  $\text{H}_2\text{O}$  and  $\text{H}_2$  desorption peaks were shown to be the **sums** of first-order subpeaks which had reproducible activation energies and pre-exponential terms. For  $\text{H}_2\text{O}$  desorption, these peaks, of **types** labeled **B**, **C**, and **D** had activation energies of 22, 28, and 32 kcal/mol. (Earlier work identified type **A** peaks with an activation energy of 18 kcal/mol.) Enhancement of desorption of  $\text{H}_2\text{O}$  in rate and quantity by  $\text{H}_2$  in the sweep **gas** was confirmed. It appears that the effectiveness of  $\text{H}_2$  in this enhancement is not in modifying the activation energy and pre-exponential term for a given kind of site, but rather in changing the populations of sites participating in the desorption processes so that sites with lower activation energies are increasingly involved. Only C and D subpeaks were involved in  $\text{H}_2$  desorption and they had activation energies close to and perhaps only 1 kcal/mol higher than the analogous peaks for  $\text{H}_2$  evolution.

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9.1 MICROSTRUCTURE OF $Al_2O_3$ AND $MgAl_2O_4$ PREIMPLANTED WITH H, He, C AND IRRADIATED WITH $Ar^+$ IONS (Japan Atomic Energy Research Institute, Oak Ridge National Laboratory and Harwell Laboratory) .....	293
<p>Polycrystalline spinel (<math>MgAl_2O_4</math>) and alumina (<math>Al_2O_3</math>) were irradiated under three, different sets of ion irradiation conditions in order to examine the effect of displacement damage and transmutation product formation on their microstructural evolution. Specimens were implanted with H, He and C alone, or irradiated with 4 MeV <math>Ar^+</math> ions alone, or preimplanted with H, He, C and subsequently irradiated with 4 MeV <math>Ar^+</math> ions. Irradiated specimens were observed with transmission electron microscopy following irradiation to damage levels of 0.6 to 30 dpa at 260 to 630°C for spinel and 0.6 to 18 dpa at 260 to 810°C for alumina. In both spinel and alumina, no radiation effect were observable in specimens implanted with H, He and C alone, and the preinjected transmutation products did not have a significant effect on dislocation loop size or density in <math>Ar^+</math> irradiated specimens. In alumina, cavities were formed in all of the observed <math>Ar^+</math> irradiated specimens for both the preimplanted H, He and C case and the <math>Ar^+</math> alone case. In spinel, cavity formation was only observed in preimplanted specimens irradiated with <math>Ar^+</math> ions at relatively high temperatures and/or doses.</p>	
9.2 MICROSTRUCTURAL CHANGES IN OXIDE CERAMICS FOLLOWING LIGHT ION IRRADIATION (Oak Ridge National Laboratory) .....	302
<p>Irradiation of ceramics such as <math>Al_2O_3</math> and <math>MgAl_2O_4</math> with energetic light ions causes a suppression in the nucleation of dislocation loop. The results of recent electron microscope studies on ion irradiated oxide ceramics are summarized. It is proposed that the anomalous suppression of dislocation loop formation may be associated with the high proportion of energy lost to electronic ionization events compared to displacement damage events during light ion irradiation.</p>	
9.3 PREPARATION OF $MgAl_2O_4$ SPINEL CONTAINING CONTROLLED AMOUNTS OF $^{17}O$ ISOTOPE (Oak Ridge National Laboratory) .....	310
<p>Magnesium aluminate spinel (<math>MgAl_2O_4</math>) with an <math>^{17}O</math> enrichment (<math>^{17}O/O_{TOT}</math>) of about 23 at. % was prepared by reacting fine mixtures of aluminum hydroxide (enriched with <math>^{17}O</math>) and <math>MgO</math> of normal isotopic content. The material was prepared for experiments in which the radiation damage produced in a fusion reactor is simulated by fission reactor exposures. The powder mixtures were obtained by hydrolyzing, with water containing the <math>^{17}O</math> isotope, a mixture of aluminum isopropoxide and <math>MgO</math> powder. The mixture was converted into pure spinel by a series of heat treatment and grindings. Essentially fully dense bodies, which contained about 45% of the <math>^{17}O</math> isotope initially present in the water, were successfully fabricated provided that all thermal treatment were conducted in argon or vacuum atmospheres.</p>	
9.4 IRRADIATION OF CONVENTIONAL AND ISOTOPICALLY TAILORED CERAMICS IN HFIR (Oak Ridge National Laboratory, Rensselaer Polytechnic Institute, and Los Alamos National Laboratory) .....	314
<p><math>Al_2O_3</math> and <math>MgAl_2O_4</math> specimens containing <math>^{17}O</math> isotope concentrations of 0 to 25 at. % (<math>^{17}O/O_{TOT}</math>) have been prepared for irradiation in HFIR in order to investigate the effects of helium formation on their microstructure and dielectric and mechanical properties. SiC/SiC flexure bars, ceramic fibers and a wide range of commercial ceramic TEM disks has also been included in the irradiation capsules. The specimens will be irradiated at temperatures of 100, 350, and 600°C in two irradiation capsules that will receive respective maximum neutron fluences of <math>2.4 \times 10^{26}</math> and <math>7.2 \times 10^{26}</math> n/m<sup>2</sup> (<math>E &gt; 0.1</math> MeV).</p>	
9.5 IRRADIATION OF CERAMICS IN FFTF (Oak Ridge National Laboratory and Rensselaer Polytechnic Institute) .....	317
<p>A wide range of advanced structural ceramics have been prepared and shipped to FFTF for irradiation to a nominal fluence <math>\sim 6 \times 10^{26}</math> n/m<sup>2</sup> (<math>E &gt; 0.1</math> MeV) at temperatures of 420 to 800°C. Most of the ceramics will be irradiated as TEM disks, but SiC/SiC flexural bars, electrical conductivity disks, and bundles of commercial ceramic fibers will also be irradiated in separate irradiation capsules.</p>	
9.6 CROSS SECTIONAL MEASUREMENT OF ELASTIC MODULUS FOR ION BEAM DAMAGED SILICON CARBIDE - (Oak Ridge National Laboratory and Rensselaer Polytechnic Institute) .....	320
<p>The application of a microindentation technique to measure the elastic modulus of carbon beam implanted silicon carbide is presented. Samples of Chemically Vapor Deposited (CVD) Silicon Carbide/Nicalon composites have been implanted to a damage level of 30 dpa at room temperature. The samples were then prepared in cross section and the microstructure was analyzed and modulus was measured along the damage path of the carbon ions. Both CVD silicon carbide and Nicalon silicon carbide was seen to amorphize at room temperature with the threshold for the CVD being approximately 15 dpa. Amorphization of Nicalon silicon carbide fiber was seen to occur over the entire range of the carbon path implying a much lower threshold for amorphization. Elastic moduli were seen to decrease significantly (from 440 GPa to 280 GPa) for the CVD material with the minimum modulus corresponding to the maximum damage region of the carbon beam. The modulus of Nicalon showed the opposite behavior. The modulus for the fiber was increased over most of the damaged region from an undamaged value of 170 GPa to a maximum of 210 GPa.</p>	

9.7 IN-SITU MEASUREMENT OF RADIATION INDUCED CONDUCTIVITY IN CERAMICS (Los Alamos National Laboratory) ..... 328

Recent data by E R. Hodgson<sup>1,2</sup> and by G. P. Peils<sup>3,4</sup> shows enhanced dielectric loss and electrical breakdown in ceramics during irradiation with protons and electrons and while subject to an externally applied DC electric field. We have investigated this radiation-induced conductivity effect at frequencies between 100 Hz and 10 MHz by irradiating sapphire with 3 MeV protons at 300 K. Calculations of energy deposition and displacement damage in the sapphire were made using TRIM 90 on an IBM-AT. Results show an immediate increase in loss tangent at onset of irradiation that depends on both frequency and time.

9.8 CORROSION AND ELECTRICAL PROPERTIES OF CERAMIC INSULATORS AFTER EXPOSURE TO FLOWING LITHIUM AT 400°C · (Argonne National Laboratory) ..... 330

Based on a preliminary survey of more than 15 oxides and nitrides, four ceramic materials (CaO, MgO, Y<sub>2</sub>O<sub>3</sub>, and BN) were identified as candidates for insulator coating development. These compounds were fabricated by a variety of techniques and exposed to flowing lithium at 400°C to assess chemical compatibility. Yttrium oxide exhibited excellent corrosion resistance in flowing liquid Li at 400°C; its corrosion rate was calculated to be 0.042 m/yr. Resistivity measurements were taken before and after Li exposure to determine the effects of Li on the electrical properties of Y<sub>2</sub>O<sub>3</sub>. No deterioration in the resistivity could be measured; in both cases the resistivity was >10<sup>12</sup> Ω·m. Boron nitride coatings on V-20Ti Substrates were fabricated by the newly developed ion-beam-assisted-deposition process. These coatings are expected to exhibit better compatibility with liquid Li than the previously RF-sputtered coatings. The resistivity of in-situ-formed (V,Ti)<sub>x</sub>N reaction-product layers on various V-base alloys, including V-7Cr-5Ti, V-2.5Ti-1Si, and V-20Ti, is being determined.