

DOE ER 0313 1

# ***Fusion Reactor Materials***

Semiannual Progress Report  
for Period Ending  
September 30, 1986



**U. S. Department of Energy**  
Office of Fusion Energy



DOE/ER-0313/1  
Distribution  
Categories  
UC20, 20c

**FUSION REACTOR MATERIALS  
SEMIANNUAL PROGRESS REPORT  
FOR THE PERIOD ENDING SEPTEMBER 30, 1986**

Date Published: September 1987

Prepared for  
DOE Office of Fusion Energy

Prepared by  
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MARTIN MARIETTA ENERGY SYSTEMS, INC.  
for the  
U.S. DEPARTMENT OF ENERGY  
under Contract DE-AC05-84OR21400

## FOREWORD

This is the first in a series of semiannual technical progress reports on fusion reactor materials. This report combines research and development activities which were previously reported separately in the following technical progress reports:

- Alloy Development for Irradiation Performance
- Damage Analysis and Fundamental Studies
- Special Purpose Materials

These activities are concerned principally with the effects of the neutronic and chemical environment on the properties and performance of reactor materials; together they form one element of the overall materials program being conducted in support of the Magnetic Fusion Energy Program of the U.S. Department of Energy. The other major element of the program is concerned with the interactions between reactor materials and the plasma and is reported separately.

The Fusion Reactor Materials Program is a national effort involving several national laboratories, universities, and industries. The purpose of this series of reports is to provide a working technical record for the use of the program participants, and to provide a means of communicating the efforts of materials scientists to the rest of the fusion community, both nationally and worldwide.

This report has been compiled and edited under the guidance of A. F. Rowcliffe, Oak Ridge National Laboratory, and D. G. Doran, Battelle-Pacific Northwest Laboratory. Their efforts, the work of the publications staff in the Metals and Ceramics Division at ORNL, and the many persons who made technical contributions are gratefully acknowledged. T. C. Reuther, Reactor Technologies Branch, has responsibility within DOE for the programs reported on in this document.

G. M. Haas, Chief  
Reactor Technologies Branch  
Office of Fusion Energy

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*The Workshop on Mechanical Properties of Fusion Reactor Materials was held in conjunction with the Second International Conference on Fusion Reactor Materials to encourage international representation in this workshop series. The meeting was held in Chicago, Illinois in April 1986, with participants from Japan, Europe, and the United States. The topics covered were: 1) flow processes, 2) time dependent crack growth, 3) time independent crack growth/brittle fracture, and 4) environment assisted crack growth. The background, status, and recommended research are summarized in this report for each of the workshop topics.*

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316 stainless steel, and the incubation effects are less obvious than in copper. In HT9 the 14 MeV irradiation effects are relatively small, and there is no effect of temperature for fluences up to  $2.5 \times 10^{18}$  n/cm<sup>2</sup>.

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The scale of the fluctuations at temperatures >600°C is comparable to that found in the Santa Catharina meteorite, which is roughly Fe-35%Ni in composition. This and other non-radiation data support the proposal that the Fe-Ni system in the absence of irradiation tends to spinodally decompose in the invar regime but at a very sluggish rate. Thus it appears that radiation accelerates rather than induces the decomposition of Fe-Ni and Fe-Ni-Cr invar alloys.

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The residual oxygen plays an important role in promoting void formation. Lowering the oxygen content from 180 appm to 75 appm reduces void density and increases void size remarkably.

Small amounts of helium (10 appm) enhanced the void nucleation remarkably in both high (180 appm) and low (75 appm) oxygen content samples, while larger amounts of helium (30 appm) reduced the observable void density in the high oxygen content sample.

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Surface energy values lower than those determined experimentally are often utilized in theories of void nucleation and growth in metals. Utilization of established surface energy values generally predicts no swelling in the absence of

helium. However, swelling occurs in many metals even in the absence of helium. Surface active impurities, such as oxygen, can readily account for this discrepancy by reducing the surface energy of metals. This investigation shows that very low concentrations of oxygen in nickel can achieve the necessary decrease in surface energy.

A model has been developed to calculate the requisite quantity of oxygen in solution to stabilize voids. The criterion for void stability is that the void be the most energetically stable vacancy cluster in the metal. Knowing the fraction of oxygen which chemisorbs and the surface coverage required permits the determination of initial oxygen concentration needed to promote void stability. Calculations have been performed for nickel.

The model has been tested by irradiating nickel with 14-MeV Ni ions at 500°C. Oxygen was preinjected into one sample to a concentration of 75 appm. The irradiation reached a fluence of  $3 \times 10^{20}$  Ni-ion/m<sup>2</sup> (28 dpa at the damage peak). The irradiated foils were examined in cross section in the electron microscope.

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*The tensile behavior more nearly reflected the differences in initial condition than the differences between irradiation properties of the alloys, the cold-worked material remaining stronger even after irradiation. In general, irradiation in this temperature range clearly produced hardening with accompanying reduction in ductility. All four alloys, with the possible exception of JPCA, exhibited rather low uniform elongation. There was no evidence for helium embrittlement.*

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*Solution-annealed (SA) type 316 stainless steel was irradiated to 36 dpa at 600°C in HFIR. Large irradiation-produced (100 to 270 nm) precipitates were observed in irradiated specimens and many voids (3 to 30 nm in diameter) were attached to these precipitates. In addition, many fine helium bubbles (1 to 5 nm in diameter) and dislocation loops (~5 nm) were observed inside these precipitates. Most of these precipitates were identified as M<sub>6</sub>C (eta) phase, enriched in Ni, Cr, Mo, and Si.*

6.2.3	Temperature Dependence of Swelling in Type 316 Stainless Steel Irradiated To About 33 dpa In HFIR (Japan Atomic Energy Research Institute, assigned to Oak Ridge National Laboratory and Oak Ridge National Laboratory . . . . .	280
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*Solution-annealed (SA) and 20%-cold-worked (20% CW) type 316 stainless steels were irradiated in HFIR up to 33 to 36 dpa and 2250 to 2335 appm He at irradiation temperatures of 300 and 400°C. Small bubbles (1.5 to 4.5 nm in diameter) were uniformly dispersed throughout the matrix at concentrations in the range 2 to 4 x 10<sup>23</sup> m<sup>-3</sup>. Swelling was very low (below 0.2%) in both materials. In SA materials, cavity size rapidly increased while the number density decreased at irradiation temperatures of 500°C and above. Swelling appeared to be a maximum at 500°C (>1%). Most of the cavities were voids at 600°C. On the other hand, in 20% CW specimens, cavities were much smaller, with diameters of 6 and 9 nm at 500 and 600°C, respectively. The cavity number density at both 500 and 600°C (~1 x 10<sup>22</sup> m<sup>-3</sup>) was about one order less than at 400°C. Swelling slightly increased as irradiation temperature increased, peaking at 600°C (0.3%). Inhibition of swelling by cold working was more effective at temperatures above 500°C.*

6.2.4	The Development of Austenitic Stainless Steels for Fast Induced-Radioactivity Decay (Oak Ridge National Laboratory) . . . . .	286
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*Tensile properties were determined for six Fe-Mn-Cr-C alloys that were used previously to determine the austenite-stable region in that system. The steels were tested in the solution-annealed and 20%-cold-worked conditions. When the results were compared with type 316 stainless steel, the average behavior of the five steels compared favorably with the tensile behavior of type 316 stainless steel.*

6.3	Vanadium Alloys . . . . .	291
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6.3.1	Preparation and Fabrication of Vanadium Base Alloys (Argonne National Laboratory) . . . . .	293
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*Initial phases in the production of V-Cr-Ti alloys have been completed. All steps in the process flow sheet (1) have been activated on production of the V-15Cr-5Ti ternary alloy. Sufficient material was processed to generate a 9.65 cm (3.8") diameter cast ingot weighing 13.1 kg (29 pounds). Subsequent hot extrusion provided a change in geometry to the billet to remove the cast structure and produce a rectangular shape to initiate hot rolling procedures. This extrusion has been subdivided and is currently passing through final processing operations, which include hot rolling, warm rolling and cold finishing steps. Flat stock is now being accumulated, while characterization work such as chemistry, non-destructive testing and metallography is on-going. Plans for the fabrication of the V-10Cr-5Ti alloy have been defined and the initial alloy consolidation is under way.*

*In addition, this effort provides for correlation of complimentary fabrication work on V-12Cr-5Ti and V-10Cr-10Ti alloys at Teledyne Wah Chang Albany.*

6.3.2	Development of a Vanadium-Base Alloy Structural Material (Argonne National Laboratory) . . . . .	297
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*The V-3Ti-1Si, V-5Ti, V-20Ti, and V-15Cr-5Ti alloys were irradiated at 650°C with 4.0-MeV <sup>51</sup>V<sup>++</sup> ions to 50 dpa. Voids were not visible in the irradiated alloys. Coherent precipitates were produced in the irradiated V-5Ti and V-15Cr-5Ti alloys. The precipitates were noncoherent with the matrix in the V-20Ti alloy and partially coherent in the V-3Ti-1Si alloy. The formation of coherent precipitates may contribute to the greater susceptibility of the V-15Cr-5Ti alloy, in comparison to the V-20Ti and V-3Ti-1Si alloys, to irradiation hardening and loss of ductility after neutron irradiation. A test matrix for irradiation of V-base alloys in the FFTF reactor during cycle 9 is presented.*

6.3.3	Effect of Preinjected Helium on the Response of V-20Ti Pressurized Tubes to Neutron Irradiation (Oak Ridge National Laboratory) . . . . .	302
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*Vanadium-20% titanium tubes, pressurized to stresses of 34 and 39 MPa, were irradiated in the Experimental Breeder Reactor (EBR-II) at 700°C to a fluence of 3.9 x 10<sup>26</sup> n/m<sup>2</sup>, corresponding to a displacement damage level of 22 dpa. Sections of the tubes were injected with 15 appm He prior to irradiation to determine the effect of helium on the microstructural and creep response of this*

*alloy to irradiation. It was found that helium promoted cavity formation, primarily within existing precipitates, but total swelling remained low. Under some conditions, an apparent enhanced creep deformation due to the presence of helium was found. The results suggest that the increase in creep deformation in the presence of helium may be very sensitive to stress.*

6.4 Copper Alloys . . . . . 309

6.4.1 Effects of Neutron Irradiation to 63 dpa On the Properties of Various Commercial Copper Alloys (Westinghouse Hanford Company) . . . . . 311

*High purity copper and six commercial copper alloys were neutron irradiated to 47 and 63 dpa at about 450°C in the FFTF. Immersion density measurements showed a wide range of swelling behavior after irradiation to 63 dpa. At one extreme was CuBe in the aged and tempered (AT) condition which had densified slightly. At the other extreme was 20% CW Cu-0.1% Ag which swelled over 45%. Electrical resistivity measurements of high-conductivity alloys followed trends similar to previously published results for the same alloys irradiated to 16 dpa, namely a continued reduction in conductivity with fluence which appears to relate to transmutation products and, somewhat, to void formation and defect cluster development. At 63 dpa, the electrical conductivity of zone-refined copper had decreased significantly. The reduction was to a value comparable with that of the irradiated Cu-Al25—the Al<sub>2</sub>O<sub>3</sub> dispersion strengthened alloy. Conversely, for the moderate conductivity alloy CuBe, the electrical conductivity was unaffected for irradiation greater than 16 dpa. These results of the irradiated material were compared with electrical conductivity of unirradiated alloys examined after aging for 10,000 hours.*

*The most irradiation resistant high-conductivity, high-strength copper alloy examined after 63 dpa is Cu-Al25 followed by MZC. Cu-2.0Be, only a moderate-conductivity alloy, exhibits very consistent irradiation resistant properties. Thus, Cu-Al25 and MZC appear to be acceptable candidates for high heat flux materials in fusion reactor applications.*

6.4.2 Neutron Irradiation of Copper Alloys--Phase II (Westinghouse Hanford Company) . . . . . 321

*A second generation copper alloy experiment has been designed, built, and inserted into the Fast Flux Test Facility (FFTF) for irradiation. The experiment test matrix is heavily biased toward the examination of oxide-dispersion-strengthened (ODS) copper-based alloys. This material exhibited the most irradiation resistance of the alloys examined after fast reactor irradiation to fluences up to 63 dpa. The current experiment included matrix and weldment specimens of commercial ODS alloys that were designed to be weldable.*

6.5 Environmental Effects On Structural Alloys . . . . . 323

6.5.1 Environmental and Chemical Effects On the Properties of Vanadium-Base Alloys (Argonne National Laboratory) . . . . . 325

*The susceptibility of V-15Cr-5Ti to stress corrosion cracking in water at 288°C has been evaluated by means of constant extension rate tensile (CERT) tests. The test environments include high-purity water as well as water containing controlled levels of SO<sub>4</sub><sup>2-</sup> and NO<sub>3</sub><sup>-</sup>. Strain rates from 1 x 10<sup>-6</sup> to 5 x 10<sup>-6</sup> s<sup>-1</sup> were employed, and dissolved oxygen levels ranged from <0.005 to 7.9 wppm. No stress corrosion cracking was observed under any of the test conditions. Electrochemical potential values as a function of dissolved oxygen content were obtained from V-15Cr-5Ti, Type 304 stainless steel, and platinum electrodes in the high-temperature water.*

6.5.2 Corrosion Studies In Thermally Convective Lithium: 12Cr-1MoVW Steel and Low Activation Austenitic and Ferritic Alloys (Oak Ridge National Laboratory) . . . . . 331

*Results from experiments with austenitic and ferritic steels exposed to lithium yielded further evidence for the important role of chromium reactions in the corrosion process. Preliminary indications from the analysis of weight change data for a low nitrogen Fe-Cr-Mn steel exposed to thermally convective lithium at 500°C for 2856 h revealed that the nitrogen concentration alone cannot account for the chromium reactions observed for Fe-Cr-Mn alloys in molten lithium. Additional surface analysis of standard 12Cr-1MoVW steel specimens exposed in a thermal convection loop circulating lithium between 600 and 450°C confirmed the presence of chromium enrichment at intermediate temperatures. An*

initial study of low activation ferritic steels in thermally convective lithium at 500°C was completed. Results showed low weight losses typical of standard ferritic steels exposed under similar conditions. Steels containing a significant concentration of manganese appear to be slightly less corrosion resistant than one containing only 0.02% wt % of this element.

6.5.3 Long-Term Corrosion of Type 316 Stainless and 12Cr-1MoVW Steels In Thermally Convective Pb-17 at. % Li (Oak Ridge National Laboratory) . . . . . 337

During the current reporting period, long-term (10,000 h) baseline thermal convection loop experiments with austenitic and ferritic steels exposed to Pb-17 at. % Li were completed. The 500°C data confirmed the aggressiveness of the lead-lithium environment. Surface analysis revealed uniform attack of 12Cr-1MoVW steel with minimal change in surface composition. In contrast, the type 316 stainless steel suffered irregular attack and preferential dissolution of nickel and chromium. These observations are in accord with a model that predicts such nonuniform attack under conditions of selective leaching. Mass transfer profiles revealed that, in the case of type 316 stainless steel, the maximum deposition was not at the coldest point in the loop. Such behavior can be attributed to the product of thermodynamic and kinetic factors that vary oppositely with respect to temperature.

6.5.4 Corrosion of Ferritic Steels and V-15Cr-5Ti Alloy In Flowing Lithium (Argonne National Laboratory) . . . . . 343

Corrosion data are presented for several ferritic steels and the V-15Cr-5Ti alloy in a flowing lithium environment. The dissolution rates of low-activation ferritic steels and V-15Cr-5Ti alloy are compared with those for HT-9 and Fe-9Cr-1Mo steels. The influence of nitrogen content in lithium on the corrosion behavior of these alloys is discussed.

6.5.5 Corrosion of Ferrous Alloys In Flowing Pb-17Li Environment (Argonne National Laboratory) . . . . . 349

Corrosion data have been obtained on low-activation ferritic steels in flowing Pb-17 at. % Li environment at temperatures of 482 and 371°C. The results are compared with the dissolution behavior of ferritic HT-9 and Fe-9Cr-1Mo steels and the austenitic Type 316 stainless steel.

7. SOLID BREEDING MATERIALS . . . . . 353

7.1 The Thermal Conductivity of Mixed Beryllia/Lithium Ceramic In Sphere-Pac Form (Argonne National Laboratory) . . . . . 355

Lithium-containing ceramic tritium-breeder materials have been envisaged to be deployed within the blanket region of a fusion reactor in several possible configurations. One of these is the sphere-pac configuration. For this configuration an important material parameter is its thermal conductivity ( $K_{SP}$ ). It is well known that  $K_{SP}$  demonstrates rather complex behavior as a function of temperature, gas pressure, gas composition, particle size, and packing fraction. The interrelationship of these parameters has been satisfactorily accounted for with a hierarchical effective media theory (HEMT). For tritium self sufficiency, most lithium ceramic breeder materials would require the presence of a neutron-multiplier (e.g., Be or BeO). Here, the influence of configuration on  $K_{SP}$  (i.e., how one put the different solid components together in the sphere-pac bed) becomes important. Using a generalized HEMT (i.e., a model with capability to describe systems with more than one solid material component), we have analyzed in detail the configurational dependence of  $K_{SP}$  for sphere-pac beds composed of lithium ceramic/BeO microspheres. Substantial improvements in  $K_{SP}$  can be achieved if a configuration of lithium ceramic spheres coated with BeO is chosen. Increases in  $K_{SP}$  would lead to enhanced mechanical and thermal performance of the breeder materials.

7.2 Beatrix Materials Exchange In the International Community (Argonne National Laboratory, US Department of Energy, and CEN/Saclay) . . . . . 359

The BEATRIX experiment is an IEA-sponsored effort that involves the exchange of solid breeder materials and shared irradiation testing among research groups in several countries. The materials will be tested in both closed capsules (to evaluate material lifetime) and opened capsules (to evaluate purge-flow tritium recovery). Pre- and post-irradiation measurement of thermophysical and mechanical properties will also be carried out.

7.3 Adsorption, Dissolution, and Desorption Characteristics of the  $\text{LiAlO}_2\text{-H}_2\text{O}$  System (Argonne National Laboratory) . . . . . 362

*Experimental measurements are being made of surface adsorption of  $\text{H}_2\text{O}$  on  $\text{LiAlO}_2$ , the solubility of hydroxide in  $\text{LiAlO}_2$ , and the kinetics of release of  $\text{H}_2\text{O}$  from  $\text{LiAlO}_2$ . Up to about  $500^\circ\text{C}$ , evolution of  $\text{H}_2\text{O}$  is first order in dissolved protons (hydroxide). At higher temperatures, the reaction appears to shift to second order. Solubility of hydroxide appears to decrease with increasing temperature. A second condensed phase can appear at about  $315^\circ\text{C}$  for a partial pressure of  $\text{H}_2\text{O}$  of 550 vppm. The critical partial pressure of  $\text{H}_2\text{O}$  to form a  $\text{LiOH}$ -rich second phase can be similar for all breeders. Two different kinds of lattice sites appear to be involved in hydroxide dissolution. Surface adsorption of molecular oxygen or hydrogen can be understood to influence tritium release rates markedly; the thermodynamic and kinetic effects of these gases on the release rates operate in the same direction. "Tritium" diffusion is to be identified with triton diffusion.*

7.4 Solid Breeder Materials Fabrication and Mechanical Properties (Argonne National Laboratory) . . . . . 367

*The preliminary measurements of mechanical properties of lithium oxide were completed. Several batches of lithium zirconate ( $\text{Li}_2\text{ZrO}_3$ ) powder were synthesized and sent to Hanford Engineering and Development Laboratory to be fabricated into FUBR-1B replacement capsules. Lithium oxide powder was prepared and the fabrication of ring-shaped  $\text{Li}_2\text{O}$  samples was initiated for the CRITIC experiment. Mechanical properties tests were initiated on  $\text{LiAlO}_2$  and  $\text{Li}_2\text{ZrO}_3$ . Approximately 4000 kg of sintered lithium carbonate blocks were prepared and sent to Japan for neutronics cross-section tests.*

7.5 Tritium and Helium Retained In Fast Neutron Irradiated Lithium Ceramics As Measured By High Temperature Vacuum Extraction (Westinghouse Hanford Company) . . . . . 368

*A vacuum apparatus was designed and constructed for the rapid measurement of retained helium and tritium in lithium ceramics. The apparatus eliminated the limitations and errors associated with the acid dissolution technique and the previous vacuum annealing technique (below the melting point) and allowed more accurate and less expensive analysis techniques. Tritium retention in  $\text{Li}_2\text{ZrO}_3$  was significantly less than in the other ceramics. Tritium retention appears to possess a proportional dependence to burnup. A review of available models reveals that none fully described the absolute magnitude or the relationship of retention to temperature or burnup so that a model which considers irradiation effects is desired.*

7.6 Time Dependent Analysis of In Situ Tritium Release Curves From The Vom-22H Experiment (Westinghouse Hanford Company) . . . . . 376

*An analysis method was developed which allows the transient response of in situ tritium recovery experiments after temperature change steps to be used in calculation of diffusion coefficients. The papers by Kurasawa et al. on the conclusions reached from this work are available in the open literature and will not be reproduced here.*

7.7 The Fubr-1B Experiment And Beatrix (Westinghouse Hanford Company) . . . . . 380

*The first insertion of two subassemblies has completed its irradiation in December 1986. This irradiation exposed  $\text{Li}_2\text{O}$  and  $\text{LiAlO}_2$  to not only high temperatures but also large temperature gradients which are expected in fusion blankets. In addition, it included other materials such as  $\text{Li}_2\text{ZrO}_3$ ,  $\text{Li}_4\text{ZrO}_6$ ,  $\text{Li}_4\text{SiO}_4$  and  $\text{LiAlO}_2$  (spheres and large grain size) some of which will go to high burnups. The second insertion will contain lithium ceramics from Saclay, France; Casaccia, Italy; Karlsruhe, Federal Republic of Germany; Springfield Laboratories, England; and JAERI, Japan.*

7.8 Lithium Transport Within Closed Irradiation Capsules Containing Lithium Ceramics (Westinghouse Hanford Company) . . . . . 384

*Lithium was transported within the FURB-1A capsules which contained  $\text{Li}_2\text{O}$  and  $\text{Li}_4\text{SiO}_4$ . The temperature and lithium burnup dependence, along with the absolute magnitude of this transport, suggest that it was caused by the formation of  $\text{LiOT}$  gas above the  $\text{Li}_2\text{O}$  pellet. Although transport in  $\text{Li}_2\text{O}$  blanket designs with high temperature purge channels can produce extensive material transport, the transport within the blanket may be limited by the localized burnup of the lithium.*

7.9 Pellet Integrity And Swelling Of Lithium Ceramics (Westinghouse Hanford Company) . . . . . 389

*Differences in the pellet integrity of lithium ceramics irradiated in the EBR-II reactor were observed to be related to the level of thermal strains within the ceramics which resulted from differences in thermal conductivity and thermal expansion of the solids. Swelling in  $Li_2O$  was found to be significantly greater than that of  $Li_2ZrO_3$ ,  $LiAlO_2$ , and  $Li_4SiO_4$  at high temperatures. At  $500^\circ C$ ,  $Li_2O$  exhibited axial shrinkage which resulted in overall volumetric shrinkage of the pellets which is not presently understood. The high temperature swelling of  $Li_2O$  is thought to be caused by the high helium retention in this solid.*

7.10 The Effect of Irradiation On The Thermal Conductivity of Lithium Ceramics (Westinghouse Hanford Company) . . . . . 397

*An apparatus for measuring the thermal conductivity of irradiated lithium ceramics to  $900^\circ C$  was designed, fabricated, and tested. Special attention was necessary in order to accommodate tritium released during the high-temperature measurements.*

8. CERAMICS . . . . . 399

8.1 In-Waveguide Measurements of MMW Dielectric Properties of Candidate Fusion Ceramics (Los Alamos National Laboratory) . . . . . 401

*The "rf window" in the first-wall structure of an MFE reactor is a crucial component for introducing powerful MMW beams into the plasma for electron cyclotron resonance heating (ECRH). As a follow-up to our prior findings of serious neutron-irradiation-induced damage to the MMW dielectric properties of polycrystalline alumina and beryllia for such windows, unirradiated specimens of silicon nitride and aluminum oxynitride ("ALON") from US and Japanese sources were machined and inserted into WR-10 waveguide for computerized measurement of  $k$  and  $\tan\delta$  from 90 to 100GHz. The ALON was found to have a dielectric loss factor  $k\tan\delta$  of 0.0035-comparable to that for the alumina of last year's work. Its spinel-type structure is known to resist swelling and other mechanical property damage. A low-loss form of hot-pressed silicon nitride was also discovered.*

*Other progress includes computerized data reduction, and calculation of a correction factor yielding slightly smaller values of  $k$  and  $\tan\delta$  than reported in last years' SPM Progress Report on alumina and beryllia. Such steps are important: the in-waveguide approach at 100 GHz tolerates the small-sample requirements of fission-reactor irradiation studies (free-space techniques do not) and facilitates data collection over a wide frequency band.*

8.2 Properties and Radiation Resistance of the Candidate RF Window Materials SiC and  $Al_2O_3$  (Los Alamos National Laboratory and Osaka University) . . . . . 406

*RF windows must withstand transmission of an intense beam of electromagnetic energy. If absorption is excessive as a result either of intrinsic lossiness or degradation upon exposure to the operating environment, the resulting thermal stresses can cause structural failure. Samples of two candidate materials for this application, Hitaceram SC-101 SiC and AD-995  $Al_2O_3$ , are currently being irradiated with 14 MeV neutrons at ambient temperature in KTNS-II; at this writing a fluence of  $4 \times 10^{22}$  n/m<sup>2</sup> has been attained. Work to date has shown that in unirradiated form the SiC exhibits low transmissivity but moderately high reflectivity at  $10^{11}$  Hz, leading to consideration of its use as a mirror in RF heating systems. The alumina has enough transmissivity before irradiation to qualify this ceramic for further study as a window material under moderate irradiation conditions.*

8.3 On Neutron-Induced Damage to the Millimeter-Wave Dielectric Properties of Alumina (Los Alamos National Laboratory and Massachusetts Institute of Technology) . . . . . 408

*We report the following findings concerning the previously reported doubling of the dielectric loss factor measured (post-radiation) at 90-100GHz and room temperature for Coors AD995 alumina irradiated to an averaged fluence of  $0.95 \times 10^{26}$  n/m<sup>2</sup> ( $E > 0.1MeV$ ) at  $385^\circ C$ , in connection with the potential use of alumina as an rf-window material at ECRH frequencies:*

*\*There is some evidence such a doubling may be relatively independent of frequency regions where strong dielectric dispersion is lacking.*

*\*Associated with this doubling is a dense network of dislocation loops, apparently interstitial (as evidenced by TEM), and lattice strain in the basal planes (as evidenced by neutron diffraction).*

*\*No evidence was found for colloidal aluminum formation.*

*\*Possible mechanisms for the doubled dielectric loss include electromagnetically-induced vibration of dislocation entities, and interactions with point defects produced by displacement and transmutation events. Any future specification for a useable fusion ceramic at ECRH or lower frequencies (as for ICRH) will have to include more than a tabulation of desired property values, since compositional, microstructural, and processing variables are also important. These include the relative concentrations of major impurities, firing temperature, and cooling rates after firing.*

9. SUPERCONDUCTING MAGNET MATERIALS . . . . .	417
9.1 Irradiation Effects of Organic Insulators (National Bureau of Standards) . . . . .	419

*An integrated approach has been developed for rapid screening of the influence of component variables on the performance of electrical insulators required for superconducting magnets in magnetic fusion energy systems. It incorporates an efficient method of specimen production in the form of 3.2-mm (0.125-in) diameter rods. Test methods include short-beam shear, fracture strength ( $G_{IC}$ ), and strain-controlled torsion. The torsion test induces failure between the fiber and matrix, which is expected to be the dominant failure mode induced by cryogenic irradiation. In addition to providing quantitative data on the modulus of rupture and of rigidity, the strain control feature facilitates analysis of the stress-displacement curve in the region where damage is occurring, providing useful information on how the various component and irradiation parameters are influencing the failure mode. The torsional test facility is easily constructed, provides rapid specimen turnaround, and has a low consumption of cryogens. A specimen subjected to a torsion test may be subsequently tested by the short-beam and fracture strength methods, enabling five tests to be performed on the same specimen. Glass-fiber reinforced specimens having three types of epoxy matrix and one type of polyimide matrix have been produced and submitted to ORNL for irradiation in the NLTNIF facility, after which they will be returned to NBS for testing by these methods.*