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experiment in EBR-II, show substantially more strengthening and a different temperature range of swelling. The effects of irradiation temperature, neutron flux, and neutron spectra appear to be larger than those of the compositional variations studied. Transmission electron microscopy (TEM) examination is in progress to determine the origin of the observed changes.

- 5.2 Radiation-Induced Spinodal-Like Decomposition of Fe-Cr-Ni Alloy and Its Influence on Electropolishing of Microscopy Disks—J. M. McCarthy and F. A. Garner (Pacific Northwest Laboratory) 73

Scanning electron microscopy was used as a tool to investigate the dependence on nickel content of radiation-induced spinodal-like decomposition in Fe-15Cr-XNi alloys. It appears that the relative resistance to swelling observed in the Invar compositional regime at 510°C is concurrent with spinodal-like decomposition. The influence of this phenomenon in prolonging the loop-dominated phase of dislocation evolution and also suppressing the onset of accelerated void growth has not yet been determined.

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- 5.5 Microstructural Evolution of Neutron-Irradiated Fe-Cr-Ni Alloy at 495°C in Response to Changes in He/dpa Ratio—J. F. Stubbs and J. E. Nevling (University of Illinois), F. A. Garner (Pacific Northwest Laboratory), and R. L. Simons (Westinghouse Hanford Company) 93

A series of three Fe-15Cr-XNi alloys in both annealed and cold-worked conditions was irradiated in the Fast Flux Test Facility at 495°C at 14 dpa. The experiment was developed to determine the separate and synergistic effects of nickel and phosphorus content, cold-work, and helium/dpa ratio. This experiment was conducted without introducing variations in displacement rate, a variable known to strongly influence microstructural evolution. Each alloy condition was irradiated in two variants, one with natural nickel and one enhanced with the ⁵⁸Ni isotope. The latter variant produces helium/dpa ratios typical of fusion reactor spectra, while the former yields a much lower level of helium. The results show that helium alters the microstructural evolution somewhat at 495°C, but its effect is relatively small compared with the influences of the other variables studied. Increases in starting dislocation density, nickel content, or phosphorus level all retard swelling temporarily, while higher rates of helium generation, usually, but not always, accelerate swelling. Phosphorus addition of 0.04 wt % not only decreased swelling but led to refinement of dislocation loop microstructure and stabilization of dislocation networks created by cold working. Phosphide precipitates did not form at this temperature and dose level.

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In previous work, the phase stability of Fe-Cr-Mn alloys during irradiation was investigated in a study that included simple binaries, simple ternaries, and commercially produced alloys. These low-activation alloys are being considered for fusion reactor service in the first wall and in other structural applications subject to high neutron doses. In addition to phase instabilities observed within the grains, grain boundaries were susceptible to varying levels of precipitation dependent upon alloy composition, displacement dose, and irradiation temperature. This report describes the grain boundary microstructures that developed in these Fe-Cr-Mn alloys during irradiation.

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Fe-15Cr-25Ni austenitic alloys with various phosphorus contents were irradiated with fast neutrons in the EBR-II reactor at temperatures ranging from 399 to 649°C and between 8.2 and 14.3 dpa. Observations of microstructures and microchemical analyses were carried out to determine the various roles of phosphorus. At low irradiation temperatures and lower phosphorus contents, where no precipitate formation was observed, the phosphorus remained in solution and had a strong but variable influence on swelling and void density. However, the results suggest that more than one mechanism involving phosphorus-point defect interaction was operating and that the net effect was a result of the competition of several mechanisms. Phosphide precipitates were observed to form at higher irradiation temperatures and phosphorus levels. The formation of these precipitates then exerted a further influence on the void density and distribution.

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Martensitic/ferritic 9Cr-1MoVNb and 12Cr-1MoVW steels doped with up to 2 wt % Ni have up to 450 appm He after HFIR irradiation to ~38 dpa, but only 6 appm He after 47 dpa in FFTF. No fine helium bubbles and few or no larger voids were observable in any of these steels after FFTF irradiation at 407°C. By contrast, many voids were found in the undoped steels (30-90 appm He) irradiated in HFIR at 400°C, while voids plus many more fine helium bubbles were found in the nickel-doped steels (400-450 appm He). Irradiation in both reactors at 400°C produced significant changes in the as-tempered lath/subgrain boundary, dislocation, and precipitation structures that were sensitive to alloy composition, including doping with nickel. However, for each specific alloy the irradiation-produced changes were exactly the same, comparing samples irradiated in FFTF and HFIR, particularly the nickel-doped steels. Therefore, the increased void formation appears solely due to the increased helium generation found in HFIR. While the levels of voids are relatively low after 37 to 39 dpa in HFIR (0.1-0.4%), details of the microstructural evolution suggest that void nucleation is still progressing, and swelling could increase with dose. The effect of helium on voids remains a valid concern for fusion application that requires higher-dose experiments.

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Impact specimens of 9Cr-1MoVNb and 12Cr-1MoVW steels were irradiated in the High Flux Isotope Reactor (HFIR) at 300 and 400°C to as high as 42 dpa. Irradiation caused large increases in the ductile-brittle transition temperature (DBTT) of both steels, with the increase being greater at 400°C than at 300°C. At 400°C, shifts in DBTT of 204 and 242°C were observed for the 9Cr-1MoVNb and 12Cr-1MoVW steels, respectively. These are the largest shifts ever observed for these steels and are attributed to the higher helium concentration generated during irradiation in HFIR.

6.1.4	The Fracture Toughness Data Base for HT9 and Modified 9Cr-1Mo Irradiated in Several Reactors up to ~100 dpa—F. H. Huang (Westinghouse Hanford Company), and M. L. Hamilton (Pacific Northwest Laboratory)	161
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A summary is presented of the entire fracture toughness database for HT9 and 9Cr-1Mo generated at Hanford. Fracture toughness tests were recently performed on miniature specimens of HT9 and modified 9Cr-1Mo irradiated in FFTF to exposures ranging from 35 to about 100 dpa. At test temperatures ranging from 20 to 430°C, values of toughness and tearing modulus at 35 to 100 dpa were no lower than those obtained

previously in tests conducted on specimens irradiated in EBR-II, despite differences in orientation between the EBR-II and FFTF specimens.

- 6.1.5 Effect of Specimen Size on the Upper Shelf Energy of Ferritic Steels—A. S. Kumar (University of Missouri-Rolla) and F. A. Garner and M. L. Hamilton (Pacific Northwest Laboratory) 177

A previous effort led to the development of size effect correlations for the ductile-brittle transition temperature (DBTT) and upper shelf energy (USE) of ferritic steels. An improved methodology is proposed that can be used to better predict the USE based on subsized specimen data. The proposed methodology utilizes the partitioning of the USE to energies required for crack initiation and crack propagation. Notched-only Charpy specimens are used in conjunction with precracked specimens to separate the two components. An unirradiated ferritic steel, HT-9, was used to demonstrate the validity of the methodology. Unlike previous correlations that were limited in their applicability to either highly ductile or brittle material, the proposed methodology is expected to be applicable over a wide range of ductility and to be particularly useful for materials which harden significantly during irradiation.

- 6.1.6 Processing of Two Iron-Chromium Oxide Dispersion Strengthened Steels by Mechanical Alloying—A. N. Niemi, M. G. McKimpton (Michigan Technology Institute), and D. S. Gelles (Pacific Northwest Laboratory) 187

Two low-activation ferritic ODS alloys have been manufactured, using mechanical alloying procedures, into extruded bar. The alloy compositions in weight percent are: Fe-14Cr-1.0Ti-0.5W-0.25Y₂O₃ and Fe-9Cr-2.0W-0.3V-0.08C-0.25Y₂O₃. Dispersoid phase instability is indicated in the Fe-9Cr carbon-containing alloy, but the 14Cr alloy appears to offer a novel material which may be suitable for first wall applications and warrants further study.

- 6.1.7 Microstructural Examination of HT-9 Irradiated in the FFTF/MOTA to 110 dpa—D. S. Gelles (Pacific Northwest Laboratory) and Akira Kohyama (University of Tokyo) 193

HT-9 in two heat treatment conditions has been examined following irradiation at 420°C to 114 dpa. The irradiations were performed in the Fast Flux Test Facility Materials Open Test Assembly (FFTF/MOTA). Void swelling is found in both conditions, with swelling values as high as 0.9% in isolated regions. Voids show a wide range of truncation between cubic and octahedral shapes. Void swelling appears to vary as a function of preirradiation heat treatment, whereas the dislocation structure and precipitation that developed during irradiation is unaffected. Quantitative microstructural measurements are in good agreement with results on similar simple alloys.

- 6.1.8 Irradiation Creep of Ferritic (and other BCC) Alloys—R. J. Puigh (Westinghouse Hanford Company) and D. S. Gelles (Pacific Northwest Laboratory) 201

Ferritic/martensitic alloys are now being used as structural materials in several reactor systems and are being considered as structural materials for future fusion reactors. The irradiation creep response of body centered cubic (BCC) alloys has been studied for over 20 years; however, only in the last 10 years has the effort narrowed to concentrate on the irradiation creep behavior of ferritic/martensitic alloys. This paper reviews our current understanding of irradiation creep behavior in ferritic alloys by reviewing the literature and reporting new data on the topic.

- 6.2 Austenitic Stainless Steels 226

- 6.2.1 The Development of Austenitic Stainless Steels for Fast Induced-Radioactivity Decay—R. L. Klueh and P. J. Maziasz (Oak Ridge National Laboratory) 227

A program is under way to develop a nickel-free austenitic stainless steel for fusion-reactor applications. Previous work has shown that an austenite-stable alloy should be possible with a base composition of Fe-20Mn-12Cr-0.25C. Tensile properties for this base composition were comparable to those of type 316 stainless steel. To improve strength and irradiation resistance, closely controlled quantities of W, Ti, V, C, B, and P were added to this base. Such additions resulted in improved tensile properties over those for type 316 stainless steel in both the solution-annealed and 20% cold-worked conditions.

- 6.2.2 Development of Tensile Property Relations for ITER Data Base—M. L. Grossbeck (Oak Ridge National Laboratory) 243

Tensile data from the Oak Ridge Matrix (Fusion Implementing Agreement Annex III), the U.S. Japan collaboration, and the available literature were reviewed. Type 316 stainless steel, in both cold-worked and annealed

conditions and PCA (both U.S. and Japanese heats) were included. Equations were developed for yield strength, uniform elongation, and total elongation. In many cases, one expression could be used for alloys and conditions; in others, separate equations had to be used. In all cases an attempt was made to provide a conservative expression rather than to have the best fit to the data. Especially in the case of strength, the value was rather insensitive to alloy composition.

- 6.2.3 Tensile Properties of Austenitic Stainless Steels Irradiated in the ORR Spectral Tailoring Experiment ORR-MFE-6J and -7J—M. L. Grossbeck (Oak Ridge National Laboratory), T. Sawai, S. Jitsukawa (Japan Atomic Energy Research Institute, assigned to ORNL), and L. T. Gibson (Oak Ridge National Laboratory) 259

Tensile properties were found to be consistent with those of previous irradiations in mixed-spectrum reactors. The yield strength at 6PC was found to be less than that at 330°C, but this can be understood in terms of hardening by dislocation loops. The properties of welds were found to resemble those of annealed base metal whether or not the weld was made in annealed or cold-worked material.

- 6.2.4 Development of Low Activation Fe-Mn and Fe-Mn-Cr Alloys for Fusion Service—L. D. Thompson (San Diego State University) and T. Lechtenberg (General Atomics) 269

Recent attention in the fusion materials research community has been focused on attempting to define compositional requirements for low-activation stainless steel alloys for first wall and blanket structures of future fusion reactors. Alloy design efforts have been initiated by the NIM, as well as in other fusion materials programs, to develop materials and microstructures inherently resistant to neutron damage. More recently, to address the concern about expected post-service neutron activation characteristics, the baseline programs focused primarily on developing neutron damage resistant materials have expanded to include ME investigation and development of low-activation structural steels containing manganese (Refs. 1-5). Iron alloys containing 25-50% Mn and ternary alloys containing 10-20% Mn with 5-20% Cr are included in the alloy classes being studied.

The former U.S. Energy Research and Development Administration (ERDA), and now the Department of Energy (DOE), was interested in developing alternative austenitic stainless steels which would rely less on strategic and expensive alloying elements for their properties. They sponsored a substantial alloy design effort during the 1970s directed at investigating the properties of Fe-Mn and Fe-Mn-Cr alloy compositions for cryogenic applications (such as liquid natural gas containment structures) which were similar to those of interest in the fusion materials program. While the database for the Fe-Mn and Fe-Mn-Cr systems is limited, the data obtained in these earlier studies are useful in understanding the alloy design capabilities of this system and for providing guidance to the current program. Many of the compositions previously investigated complement those in the NIM program and the data we will report will help establish trends in behavior. Mechanical properties and microstructural characterization data and their correlations are presented for those alloy systems of general interest to the present NIM program and ITER alloy development efforts.

- 6.2.5 Effects of Low-Temperature Neutron Irradiation on the Properties of 300 Series Stainless Steels—G. R. Odette and G. E. Lucas (University of California, Santa Barbara) 313

Neutron irradiation of austenitic stainless steels can result in significant property changes. Property degradation appears to be greatest for irradiation temperatures (and comparable annealing temperatures) near 300°C. Here, hardening and loss of ductility may reduce fracture toughness values to as low as 45 MPa√m by exposure levels of 6 dpa. These data suggest an experimental test program designed to evaluate critical microstructural and mechanical property changes in austenitic stainless steels at low irradiation temperatures.

- 6.3 Vanadium Alloys 337

- 6.3.1 Swelling of Neutron-Irradiated Vanadium Alloys—B. A. Loomis and D. L. Smith (Argonne National Laboratory) and F. A. Garner (Pacific Northwest Laboratory) 339

The swelling of V-10.0Cr-0.1Al, V-14.1Cr-0.3Al, V-3.1Ti-0.3Si, V-4.9Ti, V-9.8Ti, V-14.4Ti, V-17.7Ti, V-20.0Ti, V-14.4Cr-0.3Ti-0.3Al, V-14.1Cr-1.0Ti-0.3Al, V-13.7Cr-4.8Ti, V-9.0Cr-3.3Fe-1.2Zr (Vanstar-7), V-14.5Ti-7.2Cr, V-8.6W, V-4.0Mo, and V-12.3Ni alloys and unalloyed V was determined after neutron irradiation at 420°C and 600°C to irradiation damage levels ranging from 17 to 77 dpa in the FFTF-MOTA reactor facility. The swelling of these alloys was obtained from a determination of ME density for the unirradiated and irradiated alloys on immersion in CCl₄. The swelling of unalloyed V at 600°C was substantially increased by the addition of Cr. The addition of either Ni, W, or Mo to V had a relatively minor effect on the swelling of V. The

swelling of the V-E-Ni alloys was strongly dependent on the Ti concentration. The swelling of IMC V-3.1Ti-0.3Si and V-14.4Ti alloys at 600°C was greater than that exhibited by the other binary V-Ti alloys. The Vanstar-7 alloy underwent larger swelling than IMC V-Ti and V-E-Ni alloys. For the binary V-Ti alloys and IMC ternary V-Cr-Ti alloys, the dependence of swelling on IMC amount of irradiation damage was <0.1% swelling per dpa.

6.3.2 Literature Review of Research on Vanadium and Vanadium Base Alloys for Use in Fusion Reactor First Wall/Blanket Applications—C. A. Marsh and A. B. Hull (Argonne National Laboratory) 347

The literature was reviewed for research on the fabrication of vanadium base alloys and the effects of chemical environment, helium implantation, and neutron irradiation on the mechanical properties, microstructure, and corrosion behavior of vanadium and vanadium base alloys. The relevant material was compiled into an annotated bibliography of more than 100 representative references. These references address the topics highlighted in this report.

6.4 Copper Alloys 349

6.4.1 Overview of Copper Irradiation Programs—F. A. Garner, M. L. Hamilton (Pacific Northwest Laboratory), K. R. Anderson, J. F. Stubbins (University of Illinois), B. N. Singh, A. Horsewell (RISO National Laboratory), and W. F. Sommer (Los Alamos National Laboratory) 351

Researchers (at Pacific Northwest Laboratory) are collaborating with scientists from RISO National Laboratory, Los Alamos National Laboratory, and the University of Illinois to generate data on the response to radiation of copper alloys intended for use in ITER, NET, and long-term fusion devices. An overview of these experiments is presented.

6.4.2 Electrical Resistivity Changes Induced in Copper Alloys by Fast Neutron Irradiation—K. R. Anderson (NORCUS Program, University of Illinois), F. A. Garner, M. L. Hamilton (Pacific Northwest Laboratory), and J. F. Stubbins (University of Illinois) 357

Thirteen copper-base alloys were irradiated in FFTF/MOTA to determine the response of various alloy classes to neutron irradiation. This effort is directed towards the selection of copper alloys to serve as high heat flux components in both near-term and long-term fusion devices. Post-irradiation measurements showed that a wide variety of responses was observed in the neutron-induced changes in electrical resistivity. Tensile tests are in progress, and microscopy examination will be initiated soon.

6.5 Environmental Effects on Structural Alloys 371

6.5.1 Design of an Electrochemical Testing System to Evaluate Sensitization of Austenitic Stainless Steels Using Miniaturized Specimens—T. Inazumi [Japan Atomic Energy Research Institute (JAERI), assigned to ORNL] and G.E.C. Bell (Oak Ridge Associated Universities) 373

An electrochemical testing system was developed to evaluate the sensitization of austenitic stainless steels using miniaturized disk-type specimens, 3 mm diam by 0.26 mm thick. The specimens are also suitable for examination using transmission electron microscopy (TEM) after electrochemical testing. The apparatus consists of a specimen holder in which a miniaturized specimen is mounted as the working electrode, a test cell designed to handle radioactive materials and waste, and a potentiostat/galvanostat. Sensitization of thermally aged austenitic stainless steel specimens was successfully detected by the single-loop electrochemical potentiokinetic reactivation (SL-EPR) method.

6.5.2 Aqueous Stress Corrosion of Austenitic Steels—H. Khalak, A. B. Hull, and T. F. Kassner (Argonne National Laboratory) 379

Stress corrosion cracking (SCC) of austenitic stainless steel in water is considered a key unresolved issue for the U.S. ITER shield and blanket design activities. A computer code has been developed to determine the concentration of radiolytic species produced in the aqueous environment in various subsystems of a fusion reactor in order to estimate the subsequent likelihood of stress corrosion cracking. This code also serves as a valuable precursor to slow-strain-rate tests to determine the SCC susceptibility of austenitic stainless steels. The code is being benchmarked with the concentrations of radiolytic molecular species in boiling water reactors. Samples of candidate steels, primarily Type 316 NGSS, are being prepared to initiate low-strain-rate tests.

6.5.3 Corrosion and Compatibility Studies in Flowing Lithium Environments—A. E. Hull and O. K. Chopra (Argonne National Laboratory)

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The Argonne Lithium Corrosion Experiment (ALICE), a high-flow-velocity ferritic steel forced circulation test facility, is now fully operational and is being used to address the effects of fluid flow velocity and nonmetallic elements on corrosion, mass transfer, and deposition in liquid metal systems. Hydrogen distribution data were obtained from the austenitic steel Fatigue and Failure Testing in Lithium (FFTL-3) facility in which the low circulation velocity provided only for impurity control of the liquid metal; analysis of these data indicates that hydrogen fractionates between lithium and vanadium alloys in accordance with the thermodynamic distribution coefficients.

6.5.4 Assessment of Stress-Corrosion Cracking in a Water-Cooled ITER—R. H. Jones and S. M. Bruemmer (Pacific Northwest Laboratory)

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Water-cooled, near-term reactors will operate under conditions at which SCC is possible; however, control of material purity and processing and coolant chemistry can either eliminate or greatly reduce the probability of this type of structural failure. This evaluation has focused on an assessment of water impurity effects on SCC of austenitic stainless steel at temperatures below 100°C and on the conditions controlling sensitization in the fusion heat of Type 316 SS and the fusion materials heat of modified Type 316 SS designated as PCA. This assessment identifies the dominant effect of small concentrations of impurities in high-purity water on SCC such that crack growth rates at 25–75°C in water with as little as 6–16 ppm Cl⁻ are equal to the crack growth rates at 200–300°C in high-purity water. These effects are primarily for sensitized Type 304 SS, so analysis of sensitization behavior of fusion austenitic alloys was also undertaken. An SSDOS model developed at PNL was used to make these assessments, and correlation to experimental results for Type 316 SS was very good. Both the fusion heat of Type 316 SS and PCA can be severely sensitized, but with proper thermal treatment, it should be possible to avoid sensitization.

7. SOLID BREEDING MATERIALS

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7.1 A Fast Neutron, In Situ Tritium Recovery Experiment on Solid Breeder Materials—G. W. Hollenberg (Pacific Northwest Laboratory), T. Kurasawa and H. Watanabe (Japan Atomic Energy Research Institute), S. E. Berk (U.S. DOE/OFE), Institute), I. J. Hastings and J. Miller (Atomic Energy of Canada Research Company), D. E. Baker, R. E. Eauer, and R. J. Puigh (Westinghouse Hanford Company)

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An in situ tritium recovery experiment has been designed and is being fabricated for the irradiation of Li₂O in the Fast Flux Test Facility (FFTF). Two in situ tritium recovery canisters will be irradiated with lithium atom burnups to 4%. One canister will provide fundamental data on tritium release as a function of temperature, gas composition, and flow rate. The other canister will contain solid pellet specimens with large (430°C) radial temperature gradients in order to provide integrated performance data.

7.2 Interfacial Roughness and the Thermal Conductivity of a Sphere-Pac Bed—S. W. Tam and C. E. Johnson (Argonne National Laboratory)

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Various configurations for the solid breeder (e.g., Li₂O) plus neutron multiplier (i.e., Be) component has been considered for blanket application in fusion reactor technology. One possible option is the sphere-pac configuration. This consists of the solid component in the form of small particles (ideally as spheres) put together in a bed in a close-packed manner. In order to achieve high packing density, particles of several different sizes need to be used. The interstitial spaces between particles of a given size are packed with particles of the next smaller size and so on in a hierarchical manner. The packing process can be achieved by vibratory compaction. Much experience has been gained on this configuration in fission reactor related technology. In particular extensive data has been gathered via fission-related research as well as from high thermal insulation technology work on the thermal conductivities K_{sp} of such systems. The heat conduction properties of systems possess unusual characteristics due to the convoluted interweaving bicontinuous nature of the solid component and the porosity phase. For example, when sphere-pac beds are immersed in a gas (e.g., He, Ar), the thermal conductivities K_{sp} are found to be dependent on the gas pressure P. Typically, K_{sp} increases rapidly by 50–100%, when P is raised by as little as a MPa and then changes much more slowly thereafter (see Fig. 1 and discussion in the following section). It may be thought that such subtle behavior may be exploited to the advantage of blanket technology if a sphere pac configuration is utilized in the breeder-plus-multiplier component. However, straightforward extrapolation from fission and thermal and insulation technological experience to a fusion blanket environment should be treated with care since quite different materials characteristics are involved. It turns out that a careful analysis of the physics of the situation reveals that a gas pressure-

sensitive thermal conductivity is a much less prominent effect than would have been suggested from other technology experiences alone. In order to understand this, one needs to first appreciate the reasons why such pressure effects are so dramatic in fission and thermal insulation materials.

7.3 Tritium Transport Modeling—J. P. Kopasz and C. E. Johnson (Argonne National Laboratory) 411

The modeling effort for this period was focused on the desorption process. The desorption process consists of a surface reaction and desorption of surface-bound tritium in the form of HTO, T₂O, HT, or T₂. Of concern are the order of the surface reaction and the energetics of the surface reaction and desorption step. In most of our previous work, desorption of tritium was considered to be first order in tritium due to the expected excess of hydrogen in the system. Equations for desorption which are second order in tritium were derived for situations where the hydrogen concentration in the breeder surface would be low and comparable to the expected tritium concentration. Expressions for tritium inventory were derived and calculations were performed to determine the grain radius regimes where desorption would be expected to be the rate controlling release process. In our previous work we have found evidence that the desorption energetics change as a function of surface coverage. We believe that this is due to the presence of multiple sites for desorption of tritium from the ceramic. To support this view we have analyzed constant-temperature heating tritium release experiments from the literature to determine the number of desorption sites and to obtain estimates of the desorption activation energies. Estimates of activation energies for desorption of tritium from Y₂O₃ and Li₄SiO₄ were obtained.

7.4 Adsorption, Dissolution, and Desorption Characteristics of the LiAlO₂-H₂O(g) System—Albert K. Fischer and Carl E. Johnson (Argonne National Laboratory) 419

Adsorption of H₂O(g), dissolution of OH⁻, and rates of evolution of H₂O(g) are being measured for the LiAlO₂-H₂O(g) system. These thermodynamic and kinetic data for these processes relate to the issues of tritium retention and release, and, hence, to concerns about tritium inventory in ceramic tritium breeder materials. The information will enable (1) comparison of candidate breeder materials, (2) calculation of operating conditions, and (3) elucidation of the principles underlying the behavior of tritium in breeder materials.

8. CERAMICS 423