FOREWORD iii

1.0 IRRADIATION FACILITIES, TEST MATRICES, AND EXPERIMENTAL METHODS 1

1.1 FABRICATION OF CREEP AND SWELLING SPECIMENS FOR MOTA-2B IRRADIATION - (Westinghouse Hanford Company and Pacific Northwest Laboratory) 3

Forty-five pressurized tubes and five open tubes have been made from four ferritic alloys. The alloys include a simple binary alloy, Fe-9Cr, two low activation ferric alloys in the Fe-9Cr range and one low activation alloy in the 11Cr range. Specimens were made for irradiation in the Materials Open Test Assembly (MOTA), starting in cycle 12 of the Fast Flux Test Facility (FFTF), and for corresponding thermal controls. This report documents how the specimens were made and defines the pre-test geometries.

1.2 EFFECT OF MOTA SHUTDOWN PROCEDURE ON RADIATION-INDUCED MICROSTRUCTURAL EVOLUTION - (Pacific Northwest Laboratory, Westinghouse Hanford Company, University of Tokyo and University of Missouri-Rolla) 8

From the standpoint of MOTA operation, reactor scrams are the optimum method of ending an irradiation sequence. However, this procedure induces thermal stresses in core components that shorten their life. Thus, scrams are usually reserved for safety-related considerations only. Of the other shutdown procedures available, one involving helium purging of the MOTA can lead to substantial temperature reductions of the hotter canisters while the reactor is still producing power. At the end of FFTF Cycle 10C, helium purging caused a significant alteration of microstructure at the end of the MOTA 1F experiment. This procedure probably did not alter the total amount of void swelling, but most likely altered the mechanical properties due to additional hardening associated with the formation of a high density of small dislocation loops. In MOTA 1G and MOTA 2A, this purging procedure was not employed and no alteration of microstructure is anticipated.

1.3 FABRICATION AND OPERATION OF HFIR-MFE RB* SPECTRALLY TAILORED IRRADIATION CAPSULES - (Oak Ridge National Laboratory and Midwest Technical, Inc.) 17

Fabrication and operation of four HFIR-MFE-RB* capsules (60, 200,330, and 400°C) to accommodate MFE specimens previously irradiated in spectrally tailored experiments in the ORR are proceeding satisfactorily. With the exception of the 60°C capsule, where the test specimens are in direct contact with the reactor cooling water, the specimen temperatures (monitored by 21 thermocouples) are controlled by varying the thermal conductance of a small gap region between the specimen holder and the containment tube.

Irradiation of the 60 and 330°C capsules was started on July 17, 1990. As of September 30, 1991, these two capsules had completed 12 cycles of their planned 28-cycle irradiation to a damage level of 20 displacements per atom (dpa). Fabrication of parts for the other two (200 and 400°C capsules) is complete. Assembly of the 200 and 400°C capsules is scheduled for completion in FY 1993; operation of these two capsules will follow the first two (60 and 330°C).

1.4 FABRICATION AND IRRADIATION OF HFIR-MFE-JP-17, -18, AND -19 TARGET IRRADIATION CAPSULES - (Oak Ridge National Laboratory and Midwest Technical, Inc.) 30

Fabrication and irradiation of three new uninstmmented HFIR target capsules for testing 12.5-mm-diameter stainless steel fracture toughness specimens to a damage level of approximately 3 displacements per atom (dpa) at temperatures of 60-125 and 250-300°C are proceeding satisfactorily. Two low temperature capsules of identical design, designated HFIR-MFE-JP-18 and HFIR-MFE-JP-19, each contain 32 fracture toughness specimens directly cooled by reactor cooling water. Irradiation of these two capsules is nearing completion. A single helium-filled elevated temperature capsule, designated HFIR-MFE-JP-17, will contain a stack of 86 fracture toughness specimens. Additional neutronic calculations were required for this experiment to insure that it will not cause unacceptable neutron flux shifting and hot spots in HFIR fuel regions. Irradiation of this capsule is scheduled to start in late November, 1991. Included in each capsule are companion transmission electron microscopy (TEM) and SS-3 tensile specimens for correlation of microstructural, tensile, and fracture toughness properties.

1.5 AN ATTEMPT TO REDUCE RADIOACTIVITY FOR ENERGY-DISPERSIVE X-RAY ANALYSIS - (Japan Atomic Energy Research Institute) 33

A composite specimen preparation technique for the AEM was developed using unirradiated materials. The technique reduced the mass of material from a dummy irradiated specimen by more than a factor of 100. A 1-mm diam. disk was punched from a dummy irradiated 3-mm diam. transmission electron microscope (TEM) disk. The 1-mm disk was then pressed into a hole previously punched at the center of a second 3 mm diam. disk creating a composite disk. The composite disk was electropolished using a twin jet Tenupol until the thickness of the center of the composite was about 100 μm. Approximately 100 μm of nickel plating was then deposited on the surface of the thinned composite. Standard electropolishing by Tenupol unit was performed on the nickel-plated composite specimen and the composite specimen was examined by TEM.

2.0 DOSIMETRY, DAMAGE PARAMETERS AND ACTIVATION CALCULATIONS 37

2.1 HELIUM GENERATION RATES IN ISOTOPICALLY TAILORED FE-CR-NI ALLOYS IRRADIATED IN FFTF/MOTA - (Pacific Northwest Laboratory and Rockwell International) 39

Three Fe-Cr-Ni alloys have been doped with 0.4% ^{59}Ni for side-by-side irradiations of doped and undoped materials in order to determine the effects of fusion-relevant levels of helium production on microstructural development and mechanical properties. The alloys were irradiated in three successive cycles of the Materials Open Test Assembly (MOTA) located in the Fast Flux Test Facility (FFTF). Following irradiation, helium levels were measured by isotope dilution mass spectrometry. The highest level of helium achieved in doped alloys was 172 appm at 9.1 dpa for a helium(appm)-to-dpa ratio of 18.9. The overall pattern of predicted helium generation rates in doped and undoped alloys is in good agreement with the helium measurements.

2.2 HELIUM PRODUCTION RATES IN THE LASREF FACILITY - (Los Alamos National Laboratory, Rockwell International Corporation and Battelle, Pacific Northwest Laboratory) 44

Foils of Cu, Co, Fe and Ni were irradiated in LASREF. The flux was determined by measurement of the gamma ray intensities induced by transmutation. The induced helium levels were measured by an isotope dilution mass spectrometry method. For copper the heliuddpa ratio was found to be 11.1 appm/dpa, which is –40% of the rate predicted by an earlier calculation.

2.3 NEUTRON ACTIVATION ANALYSIS OF MONOLITHIC CERAMICS AND CERAMIC COMPOSITES - (Oak Ridge National Laboratory) 46

Several ceramics and ceramic composites of interest for fusion reactors have been irradiated and the isotopes responsible for the induced activity identified. Twenty six isotopes were positively identified and the activity of each measured. The two isotopes which have the combination of highest activity and the longest half life are ^{110m}Ag and ^{60}Co. The possible sources of the impurities are briefly discussed.

3.0 MATERIALS ENGINEERING AND DESIGN REQUIREMENTS. 57

3.1 MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS - (McDonnell Douglas Missile Systems Company) 59

The database effort during this period was directed towards three different program areas: 1) Material property support to ITER during the CDA phase, (2) Continuation of the Material Handbook for Fusion Energy Systems, and 3) Development of the IEA International Fusion Material Handbook.

4.0 FUNDAMENTAL MECHANICAL BEHAVIOR 61

4.1 THE INFLUENCE OF HELIUM ON MECHANICAL PROPERTIES OF MODEL AUSTENITIC ALLOYS, DETERMINED USING ^{59}Ni ISOTOPIC TAILORING AND FAST REACTOR IRRADIATION - (Pacific Northwest Laboratory) 63

Tensile testing on model Fe-Cr-Ni alloys removed from four discharges of the ^{59}Ni isotopic doping experiment in FFTF indicates that helium/dpa ratios typical of fusion reactors do not produce changes in the yield strength or elongation that are significantly different from those at much lower helium generation rates. It also appears that tensile properties approach a saturation level that is dependent only on the final irradiation temperature, but not prior temperature history or thermomechanical starting condition.

5.0 RADIATION EFFECTS: MECHANISTIC STUDIES, THEORY AND MODELING 73

5.1 THE IMPACT OF SPECTRAL EFFECTS IN FAST REACTORS ON DATA ANALYSIS AND DEVELOPMENT OF FISSION-FUSION CORRELATIONS - (Pacific Northwest Laboratory) 75

Models describing the production of freely migrating defects (FMD) during neutron irradiations were applied to materials studies in fast reactors, where neutron spectral effects have been largely ignored. Calculations for the Experimental Breeder Reactor II (EBR-II) and the Fast Flux Test Facility (FFTF) show that effects of spectral differences between in-core and out-of-core positions are significantly larger on the basis of freely migrating defects than on the basis of dpa. As an example, data on swelling behavior measured both in-core and out-of-core in EBR-II show nearly a linear dependence on FMD production but not on dpa.

5.2 NEUTRON-INDUCED SWELLING OF BINARY Ni-Al ALLOYS - (University of Illinois and Pacific Northwest Laboratory) 80

Five Ni-AI binary alloys with aluminum levels of 1.1, 2.6, 4.2, 7.1, and 13.3 weight percent were irradiated at eight temperatures ranging from 400 to 650°C in the EBR-II fast reactor. The density changes induced during irradiation were found to be strongly dependent on both aluminum content and the irradiation temperature. Dynamic solute redistribution caused by radiation-induced segregation and the temperature-dependent formation of γ' phase also appears to contribute significantly to the density changes.

5.3 PHASE SEPARATION IN FE-35Ni DURING THERMAL AGING AT 625°C - (Washington State University and Pacific Northwest Laboratory) 88

It has been proposed in other studies that the spinodal-like decomposition process observed in some helium effects studies may arise from a radiation-induced self-organization process involving segregation to helium bubbles. In this study phase separation in Fe-35Ni during thermal aging at 625°C has been investigated by means of x-ray diffraction, hardness and resistivity measurements. Line broadening of x-ray diffraction peaks demonstrates that this alloy is susceptible to thermal decomposition into two phases. Thus, helium and radiation-induced segregation are not prerequisites for decomposition to occur. However, the absence of appreciable concurrent changes in hardness or resistivity with aging time indicates that the phase separation reaction is very sluggish. Other experiments involving irradiation of Fe-35Ni and various Fe-Cr-35Ni alloys indicate that this phase separation can be accelerated substantially, however. An analysis of existing thermodynamic data has been used to demonstrate that an increased rate of spinodal decomposition can occur for the case of irradiated Fe-35Ni.

5.4 PHOSPHIDE FORMATION IN NEUTRON-IRRADIATED Fe-15Cr-25Ni-P ALLOY - (Hokkaido University and Pacific Northwest Laboratory) 99

The results of microstructural examination of irradiation-induced phosphides formed in Fe-15Cr-25Ni-0.055P during the AA-14 experiment agree with trends observed in other studies conducted in both EBR-II and FFTF. The extent of phosphide formation in the annealed condition appears to be a well defined function of both phosphorus level and irradiation temperature. It also appears to be very sensitive to the presence of cold work.

5.5 SILICON'S ROLE IN DETERMINING SWELLING IN NEUTRON-IRRADIATED Fe-Ni-Cr-Si ALLOYS - (University of Tokyo, Pacific Northwest Laboratory and University of Missouri) 104

Two silicon-modified alloy series, one based on Fe-15Cr-20Ni and another based on Fe-15Cr-25Ni were irradiated at target temperatures between **399** and 649°C in EBR-II. The influence of silicon on swelling **is** more complex than previously envisioned and indicates that silicon plays two or more competing roles while in solution. Radiation-induced formation of γ' (Ni_3Si) precipitates is dependent on silicon and nickel content, as well as temperature. Precipitation of γ' appears to play only a minor role in void formation.

5.6 ELECTRON IRRADIATION EXPERIMENTS IN SUPPORT OF FUSION MATERIALS DEVELOPMENT - (Pacific Northwest Laboratory, Hokkaido University, Tohoku University and University of Tokyo) 111

Microstructural evolution in response to 1MeV irradiation has been investigated for three simple ferritic alloys, pure beryllium, pure vanadium, and two simple vanadium alloys over a range of temperatures and doses. Microstructural evolution in Fe-3, -9, and -18Cr ferritic alloys is found to consist of crenelated, faulted a<100> loops and circular, unfaulted a/2 <111> loops at low temperatures, but with only unfaulted loops at high temperatures, The complex dislocation evolution is attributed to sigma phase precipifaults arising from chromium segregation to point defect sinks. Beryllium is found to be resistant to electron damage; the only effect observed was enhanced dislocation mobility. Pure vanadium, V-5Fe, and V-1Ni microstructural response was complicated by precipitation on heating to 400°C and above, but dislocation evolution was investigated in the range of room temperature to 300°C and at 600°C. The three materials behaved similarly, except that pure vanadium showed more rapid dislocation evolution. This difference does not explain the enhanced swelling observed in vanadium alloys.

5.7 SUMMARY OF FINAL REPORT ON THE DOE GRANT DE-FG03-84ER52110 WITH UCLA ON "RADIATION EFFECTS ON STRUCTURAL MATERIALS," - (University of California-Los Angeles) 126

Research efforts at UCLA on the effects of fusion neutron irradiation on structural materials covered a wide range of theoretical and materials engineering problems during the **six** year period supported by the DOE grant No. DE-FG03-84ER52110. In this brief report, highlights of research accomplishments are presented in the following areas:

(1) Atomic Displacements
(2) Microstructure Evolution
(3) Materials Engineering, Mechanics and Design
(4) Invention of Low-Activation Steels
(5) Research Motivated by Grant Support

6.0 DEVELOPMENT OF STRUCTURAL ALLOYS 141

6.1 FERRITIC STAINLESS STEELS 141

6.1.1 EFFECT OF IRRADIATION IN HFIR ON TENSILE PROPERTIES OF Cr-Mo STEELS - (Oak Ridge National Laboratory) 143

Tensile specimens of tempered martensitic 9Cr-1MoVNb and 12Cr-1MoVW steels with and without additions of 2% Ni were irradiated in the High Flux Isotope Reactor (HFIR) at 400 and 600°C to displacement damage levels of up to 72 dpa. Nickel was added to the steels to produce transmutation helium, and up to 700 appm He was produced in the steels with 2% Ni. Irradiation at 400°C caused an increase in strength and decrease in ductility, while irradiation at 600°C caused a decrease in strength and an increase in ductility. Results at 400°C were similar to data from previous tests on steel irradiated to ≈11 dpa **(up** to 110 appm He), where it was concluded that helium caused a strength increase in addition to the hardening caused by displacement damage alone. The present results indicate that the helium-enhanced strengthening at 400°C saturates with increasing helium concentration, similar to the saturation (at a lower strength level) with displacement-damage dose found in the absence of helium.

6.2 AUSTENITIC STAINLESS STEEL 153

6.2.1 TEMPERATURE DEPENDENCE OF THE DISLOCATION MICROSTRUCTURE OF PCA AUSTENITIC STAINLESS STEEL IRRADIATED IN ORR SPECTRALLY-TAILORED EXPERIMENTS - (Oak Ridge National Laboratory) 155

Specimens of solution-annealed (SA) and 25% cold-worked (CW) PCA austenitic stainless steel were irradiated in ORR in spectrally-tailored experiments specially designed to produce fusion-relevant He/dpa ratios (12-18 appm He/dpa). SA and CW PCA were irradiated at 330 and 400°C to 13 dpa while only CW PCA was irradiated at 60, 200, 330 and 400°C to 7.4 dpa. While cavities and fine MC precipitates were only detectable at 330 and 400°C, dislocations were a major component of the radiation-induced microstructure at 60-400°C. The mixtures of 'black-dot" loops, larger Frank loops, and network that comprised the total dislocation structure were very dependent on irradiation temperature. Both SA and CW PCA contained Frank loops and network dislocations at 330 and 400°C. Frank loop concentrations were maximum at 330°C and dislocation evolution with dose was most evident at 400°C. At 60 and 200°C, the microstructure was dominated by very dense dispersions of (1-3 nm diam.) "black-dot" loops. Some Frank loops were observed at 200°C, hut almost none were found at 60°C. Surprisingly, there was significant radiation-induced recovery of the as-cold-worked dislocation network in CW PCA at all irradiation temperatures. There appears to be a distinguishable transition in the nature of the radiation-induced microstructure between 200 and 330°C.

6.2.2 LOW-TEMPERATURE TENSILE BEHAVIOR OF IRRADIATED AUSTENITIC STAINLESS STEELS - (Japan Atomic Energy Research Institute and Oak Ridge National Laboratory) 163

The low-temperature tensile properties of type 316 stainless steel (5316) and Japanese Primary Candidate Alloy (JPCA) irradiated in spectrally tailored Oak Ridge Research Reactor (ORR) and in High Flux Isotope Reactor (HFIR) capsules are summarized. The yield and tensile strengths in the ORR-irradiated specimens were found to increase with increasing irradiation temperatures up to 330°C followed by a decrease with increasing temperature. This strength maximum was accompanied by an elongation minimum. Similar trend curves were observed in the HFIR-irradiated specimens, although the temperature of peak strength was shifted higher. The behavior can be understood roughly in terms of microstructural development of dislocation loops and recovery of network dislocations during irradiation. The temperature of peak strength seems to depend on irradiation conditions such as He:dpa ratio and irradiation dose rate.

6.2.3 THE EFFECT OF HELIUM ON MICROSTRUCTURAL EVOLUTION AND MECHANICAL PROPERTIES OF AUSTENITIC STEELS AS DETERMINED BY SPECTRAL TAILORING EXPERIMENTS - (University of Tokyo, Pacific Northwest Laboratory and University of Wisconsin-Madison) 167

Fe-15Cr-XNi alloys irradiated at both low (0.66 to 1.2) and very high (27 to 58) helium/dpa levels exhibit significantly different levels of strengthening due to an unprecedented refinement of cavity microstructure at the very high helium levels. When compounded with the nickel dependence of helium generation, the cavity distribution for some irradiation conditions and alloy compositions can be driven below the critical radius for bubble-to-void conversion, leading to a delay in swelling. The critical radius also appears to be dependent on the nickel level. The refinement may not have resulted from the high helium levels alone, however, but also may have been influenced by differences in displacement rate and temperature history in the two experiments.

6.2.4 THE COMPLEX ROLE OF PHOSPHORUS IN THE NEUTRON-INDUCED SWELLING OF TITANIUM-MODIFIED AUSTENITIC STAINLESS STEELS - (Pacific Northwest Laboratory and University of Illinois) 173

At very low levels (<0.02 wt%), phosphorus additions to austenitic stainless steels often cause a substantial increase in void swelling. This phenomenon occurs in both simple model alloys and more complex solute-modified alloys typical of materials being considered for breeder reactor or fusion reactor service. This effect is observed in both annealed and cold-worked steels. At levels above 0.02% phosphorus, there is a reversal in behavior and swelling declines. This non-monotonic behavior tends to occur at lower irradiation temperatures in relatively simple alloys, but often persists to much higher temperatures in more complex titanium-modified alloys employed in the fusion and breeder reactor programs.

6.2.5 THE STRONG INFLUENCE OF TEMPER ANNEALING CONDITIONS ON THE NEUTRON-INDUCED SWELLING OF COLD WORKED AUSTENITIC STEELS - (Pacific Northwest Laboratory, Westinghouse Hanford Company and University of Illinois) 181

Although cold working is known to strongly influence neutron-induced swelling of austenitic stainless steels, it also appears that the annealing temperature employed before cold working has an equally large effect. Two irradiation studies conducted on AISI **316** stainless steel indicate that increasing the temperature of the intermediate temper anneal leads in general to a shortening of the transient regime of swelling. The evidence is consistent with the influence of both annealing temperature and irradiation temperature on carbide formation and its subsequent influence on the microchemical evolution and swelling of the steel.

6.2.6 IRRADIATION CREEP AND CREEP RUPTURE OF TITANIUM-MODIFIED AUSTENITIC STAINLESS STEELS AND THEIR DEPENDENCE ON COLD WORK LEVEL - (Pacific Northwest Laboratory, Westinghouse Hanford Company, University of California at Berkeley and University of Missouri-Rolla) **191**

A titanium-modified austenitic type stainless steel was tested at three cold work levels to determine its creep and creep rupture properties under both thermal aging and neutron irradiation conditions. Both the thermal and irradiation creep behavior exhibit a complex non-monotonic relationship with cold work level that reflects the competition between a number of stress-sensitive and temperature-dependent microstructural processes. Increasing the degree of cold work to **30%** from the conventional **20%** level was detrimental to its performance, especially for applications about 550°C. The **20%** cold work level is preferable to the **10%** level, in terms of both in-reactor creep rupture response and initial strength.

6.2.7 DETERMINATION OF THE CREEP COMPLIANCE AND CREEP-SWELLING COUPLING COEFFICIENT FOR NEUTRON IRRADIATED TITANIUM-MODIFIED STAINLESS STEELS AT ~400°C - (University of California at Berkeley, Pacific Northwest Laboratory and Westinghouse Hanford Company) **204**

Irradiation creep data from FFTF-MOTA at **-400**°C were analyzed for nine 20% cold-worked titanium-modified type **316** stainless steels, each of which exhibits a different duration for the transient regime of swelling. One of these steels was the fusion prime candidate alloy designated PCA. The others were various developmental breeder reactor heats. The analysis was based on the assumption that the B_o + DS creep model applies to these steels at this temperature. This assumption was found to be valid. **A** creep-swelling coupling coefficient of $D \approx 0.6 \times 10^{-2}$ MPa^{-1} was found for all steels that had developed a significant level of swelling. This result is in excellent agreement with the results of earlier studies conducted in EBR-II using annealed AISI **304L** and also 10% and **20%** cold-worked AISI **316** stainless steels. There appears to be some enhancement of swelling by stress, contradicting an important assumption in the analysis and leading to an apparent but misleading nonlinearity of creep with respect to stress.

6.3 REFRACTORY METAL ALLOYS **213**

6.3.1 IRRADIATION-INDUCED PRECIPITATES IN VANADIUM **ALLOYS** CONTAINING TITANIUM - (Argonne National Laboratory) **215**

The microstructures of V-Ti, V-Cr-Ti, and V-Ti-Si alloys were characterized by transmission electron microscopy (TEM) after neutron irradiation in the FFTF at **420** and 600°C to fluences up to **114** dpa. Two types of irradiation-induced precipitates were identified, i.e., Ti_2O and $Ti_5(Si,P)_3$. Blocky Ti(O,N,C) precipitates, which form by thermal processes during ingot fabrication, **also** were observed in all unirradiated and irradiated specimens. Irradiation-induced precipitation of spherical (**<15** nm in diameter) $Ti_5(Si,P)_3$ phase was associated with superior resistance to void swelling. In specimens with negligible swelling, $Ti_5(Si,P)_3$ precipitation was significant. Irradiation-induced precipitation of the Ti_2O phase, a phase with needlelike morphology and approximately **<20** nm thick and <200 nm long, was more pronounced in high-oxygen specimens irradiated at 600°C. It seems that ductility is significantly reduced when the precipitation of Ti_2O and $Ti_5(Si,P)_3$ is pronounced. These observations indicate that initial composition; fabrication processes; actual solute compositions of Ti, **O,** N, C, P, and Si after fabrication; **O,** N, and C uptake during service; and irradiation-induced precipitation are interrelated and are important factors to consider in developing an optimized alloy.

6.3.2 THE EFFECT OF HYDROGEN, CHROMIUM, TITANIUM, AND SILICON ON THE DUCTILE-BRITTLE TRANSITION TEMPERATURE OF VANADIUM AND VANADIUM-BASE ALLOYS - (Argonne National Laboratory) 227

Ductile-brittle transition temperatures (DBTTs) were determined for dehydrogenated and hydrogenated unalloyed V and V-1Ti, V-10Ti, and V-7Cr-5Ti alloys from Charpy-impact tests. These DBTT data complement the data previously reported by Loomis et al.[1,2] on Charpy impact testing of V-3Ti-1Si, V-5Ti, V-5Cr-5Ti, V-10Cr-5Ti, V-20Ti, V-10Cr-10Ti, V-15Cr-5Ti, V-15Ti-7.5Cr, and Vanstar-7 alloys. The present experimental results show that V alloys with Ti additions (0-20 wt.%) have a minimum DBTT (≈-250°C) in an alloy containing ≈5 wt.% Ti, that addition of up to 15 wt.% Cr to V-5Ti alloy results in a substantial increase (90-160°C) of the DBTT, and that Si additions (0.25-1.0 wt.%) to V-3Ti alloy result in a significant increase (≈60°C) in DBTT. In addition, the current results show that the presence of 400-1200 appm H in unalloyed V and V alloys causes a significant increase (60-250°C) in DBTT. The DBTT of V-5Ti, V-5Cr-5Ti, V-7Cr-5Ti, and V-10Ti alloys is least affected by H. The DBTT dependence of these materials on H, Cr, Ti, and Si concentration is presented in the form of tables, graphs, and parametric equations.

6.3.3 HIGH SWELLING RATES OBSERVED IN NEUTRON-IRRADIATED V-Cr AND V-Si BINARY ALLOYS - (Pacific Northwest Laboratory, Hokkaido University, and Argonne National Laboratory) 233

Additions of 5 to 14 wt % chromium to vanadium lead to very large swelling rates during neutron irradiation of the binary alloys, with swelling increasing strongly at higher irradiation temperatures. Addition of 2 wt % silicon to vanadium also leads to very large swelling rates, but swelling decreases with increasing irradiation temperature. Addition of 1 wt % zirconium does not yield high swelling rates, however.

6.4 COPPER ALLOYS 237

6.4.1 IRRADIATION PERFORMANCE OF OXIDE DISPERSION STRENGTHENED COPPER ALLOYS TO 150 dpa AT 415°C - (University of Missouri, University of Illinois and Pacific Northwest Laboratory) 239

Results have been obtained on the post-irradiation properties of various oxide dispersion strengthened copper alloys irradiated from 34 to 150 dpa at 415°C in the Fast Flux Test Facility. The GlidCop™ alloys strengthened by Al_2O_3 continue to outperform other alloys with respect to swelling resistance, and retention of both electrical conductivity and yield strength. Several castable ODS alloys and a Cr_2O_3-strengthened alloy show increasingly poor resistance to radiation, especially in their swelling behavior. A HfO_2-strengthened alloy retains most of its strength and its electrical conductivity reaches a constant level after 50 dpa, but it exhibits a higher residual radioactivity.

6.4.2 THE JOINING OF ALLOYS FOR FUSION REACTOR APPLICATIONS - BRAZING OF COPPER-ALUMINA ALLOYS AND THE WELDABILITY OF LOW-ACTIVATION Fe-Cr-Mn-C STEELS - (Oak Ridge National Laboratory and Auburn University) 245

Induction brazed butt joints were produced from which subsize fatigue samples were fabricated. Some fatigue samples have survived over 2×10^6 cycles in a tension-compression cantilever fatigue mode.

6.5 ENVIRONMENTAL EFFECTS ON STRUCTURAL MATERIALS 251

6.5.1 SLOW STRAIN RATE TENSILE TESTING OF FUSION SPECTRALLY-TAILORED TYPE 316 MATERIAL IRRADIATED BETWEEN 60 AND 400°C IN OXYGENATED, HIGH-PURITY WATER ENVIRONMENTS - (Oak Ridge National Laboratory and Japan Atomic Energy Research Institute) 253

Specimens of a type 316 stainless steel (SS) that had been irradiated in the fusion spectrally-tailored MFE-6J/7J experiments in the Oak Ridge Research (ORR) to 7.4 dpa at temperatures of 400, 330, 200, and 60°C were SSRT tested in a recirculating autoclave system in oxygenated, high-purity water. Complete intergranular (IG) failure was observed only for the specimen irradiated at 400°C. Only 10% IG failure was observed for the 330°C irradiated specimen and no IG failure was observed for the specimens irradiated at 200 and 60°C.

8.3 HIGH-TEMPERATURE MECHANICAL AND MATERIAL DESIGN FOR SiC COMPOSITES - (University of California-Los Angeles) **284**

Silicon Carbide (SiC) fiber reinforced composites (FRC's) are strong potential candidate structural and high heat flux materials for fusion reactors. During this past decade, they have been vigorously developed for use in aerospace and transportation applications. Recent fusion reactor system studies, such as ARIES, have concluded that further development of SiC composites will result in significant safety, operational, and waste disposal advantages for fusion systems. A concise discussion of the main material and design issues related to the use of SiC FRC's as structural materials in future fusion systems is given in this paper. The status of material processing of SiC/SiC composites is first reviewed. The advantages and shortcomings of the leading processing technology, known as Chemical Vapor Infiltration are particularly highlighted. A brief outline of the design-relevant physical, mechanical, and radiation data base is then presented. SiC/SiC FRC's possess the advantage of increased apparent toughness under mechanical loading conditions. This increased toughness, however, is associated with the nucleation and propagation of small crack patterns in the structure. Design approaches and failure criteria under these conditions are discussed.

8.4 COMPOSITE MATERIALS FOR FUSION APPLICATIONS - (Pacific Northwest Laboratory) **290**

This report presents a renew and analysis of the hermetic, thermal conductivity, corrosion, crack growth and radiation damage properties of CMCs. It was concluded that the leak rates of a gaseous coolant into the plasma chamber or tritium nut of the blanket could exceed design criteria if matrix microcracking causes existing porosity to become interconnected. Thermal conductivities of unirradiated SiC/SiC and C/SiC materials are about 1/2 to 2/3 that of Type 316SS whereas the thermal conductivity for C/C composites is seven times larger. The thermal stress figure-of-merit value for CMCs exceeds that of Type 316 SS for a single thermal cycle. SiC/SiC composites are very resistant to corrosion and are expected to be compatible with He or Li coolants if the O_2 concentrations are maintained at the appropriate levels. CMCs exhibit subcritical crack growth at elevated temperatures and the crack velocity is a function of the corrosions conditions. The radiation stability of CMCs will depend on the stability of the fiber, microcracking of the matrix, and the effects of gaseous transmutation products on properties.

8.5 GAS LEAK RATE ESTIMATES FOR CERAMIC MATRIX COMPOSITES - (Pacific Northwest Laboratory) **304**

Permeation rates for He through a SiC/SiC composite have been estimated using gas flow equations for viscous and molecular flow through small orifices. The total helium leak rate was determined for a composite material, SiC/SiC, utilized as a pressure boundary between a He coolant and the plasma. The calculated flow rates were determined for the residual porosity resulting from SiC/SiC manufacture and the total first wall surface area. This leak rate was compared to previous estimates of the tolerable leak rate limits established by plasma removal rates and recycling of He in a power reactor. The calculated estimates suggest that the He leak rates are within tolerable limits if the porosity is not interconnected or is interconnected by cracks with openings no larger than 0.3 nm. Increased crack opening, induced by creep or more extensive microcracking induced by radiation or stress, could cause He leak rates to exceed the tolerable limit. These estimates suggest that the use of SiC/SiC composites for structural applications in fusion reactors will require the use of seal coatings if these materials must also perform as a pressure boundary. Similar conclusions should also apply for tritium containment.

8.6 MEASUREMENT OF THE EFFECT OF RADIATION DAMAGE TO CERAMIC COMPOSITE INTERFACIAL STRENGTH - (Oak Ridge National Laboratory and Rensselaer Polytechnic Institute) 309

SiC/C/Nicalon Composite materials have been irradiated at 300°C in HFIR to a dose of 1 dpa. Three point bend tests demonstrate a marked decrease in ultimate tensile strength while showing added composite toughness. Both strength and toughness changes are shown to be caused by a large decrease in both interfacial shear strength and interfacial friction as measured using a thin section push-out test. The degradation in interfacial properties is attributed to a debonding at the fiber/interface interface due to fiber shrinkage during irradiation.

7.0 SOLID BREEDING MATERIALS AND BERYLLIUM **263**

7.1 DESORPTION CHARACTERISTICS OF THE Li_2O SYSTEM - (Argonne National Laboratory) **265**

This reporting period is a time of transition during which temperature programmed desorption (TPD) measurements are being shifted from the $H_20(g)$-H_2-$LiAlO_2$ system to the D_2-H_2-HD-HDO-H_2O-Li_2O system.

7.2 ANALYSIS OF IN-PILE TRITIUM RELEASE EXPERIMENTS - (Argonne National Laboratory) **266**

Analysis of tritium release data from recent laboratory experiments with lithium aluminate has identified physical processes which were previously unaccounted for in tritium release models. A new model that incorporates the recent data and provides for release from multiple sites rather than only one site was developed. Calculations of tritium release using this model are in excellent agreement with the tritium release behavior reported for the MOZART experiment.

7.3 MICROSTRUCTURAL EXAMINATION OF BERYLLIUM IRRADIATED AT 500°C TO 1 DPA - (Pacific Northwest Laboratory) 268

The microstructure of commercial arc cast beryllium has been examined following irradiation in FFTF at 500°C to 1 dpa. The major effect of irradiation is the development of helium bubbles on matrix and grain boundary dislocations.

8.0 CERAMICS **271**

8.1 ANISOTROPIC DISLOCATION LOOP NUCLEATION IN ION-IRRADIATED $MgAl_2O_4$ - (Oak Ridge National Laboratory) **273**

Polycrystalline disks of stoichiometric magnesium aluminate spinel ($MgAl_2O_4$) were irradiated with **2** Mev Al^+ ions at **650**°C and subsequently analyzed in cross-section using transmission electron microscopy (TEM). Interstitial dislocation **loops** were observed on {110} and {111} habit planes. The population of **loops** on both sets of habit planes was strongly dependent on their orientation with respect to the ion beam direction. The density of loops with habit plan normals nearly perpendicular to the ion beam direction was much higher than loops with habit plane normals nearly parallel to the ion beam direction. On the other hand, the loop sue was nearly independent of habit plane orientation. This anisotropic loop nucleation does not occur in ion-irradiated metals such as copper and may be associated with the structure of displacement cascades in ceramics.

8.2 THE EFFECT OF DISPLACIVE AND IONIZING RADIATION ON THE THERMAL CONDUCTIVITY OF AL_2O_3 - (Oak Ridge National Laboratory) **277**

Recent measurements[1-3] have shown that the dielectric **loss** tangent of Al_2O_3 increases dramatically during irradiation by neutrons and protons. This large increase observed during ***in-situ*** measurements has raised the question as to the effect of radiation on the thermal conductivity under irradiation conditions. The change in the lattice component of the thermal conductivity due to several phonon scattering mechanisms is calculated. The scattering due to intrinsic phono-phonon interactions, vacancies, and to conduction band electrons is considered. The vacancies produced by displacive radiation strongly scatter phonons in the **high** frequency portion of the phonon spectrum and the resulting fractional decrease in the thermal conductivity as a function of defect concentration is calculated. Ionizing radiation produces a large increase in the number of electrons in the conduction band and these electrons will scatter phonons in the low frequency portion of the phonon spectrum. The fractional decrease in the thermal conductivity as a function of the radiation induced electrical conductivity is calculated.