



---

# Alloy Development for Irradiation Performance

Semiannual Progress Report  
For Period Ending March 31, 1982

---

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

CONTENTS

FOREWORD..		iii
1.	ANALYSIS AND EVALUATION STUDIES	1
1.1	Materials Handbook for Fusion Energy Systems (McDonnell Douglas Astronautics Company and Westinghouse Hanford Company)	2
	<p style="margin-left: 40px;"><i>Data sheets covering both the unirradiated and irradiated properties of the glass-epoxy laminate G-10 CR have been distributed to holders of the MFES. This material is used as an insulator in superconducting magnets and these data sheets represent the first step in fulfilling a commitment for the handbook to provide data on all materials, metallic and non-metallic, used in designing fusion systems. Other data sheets that should be released in the near future cover carbon, lithium compounds, and structural materials (316 SS and HT-9).</i></p>	
2.	TEST MATRICES AND METHODS DEVELOPMENT	5
2.1	Neutronic Calculations in Support of the ORR-MFE-4 Spectral Tailoring Experiments (Oak Ridge National Laboratory)	6
	<p style="margin-left: 40px;"><i>The geometric model of the ORR-MFE-4 experimental capsules has been modified, and calculated fluences are being scaled to agree with the experimental data. As of March 8, 1982, this treatment yields 36.30 at. ppm H (not including 2.0 at. ppm H from <sup>10</sup>B) and 4.35 dpa for type 316 stainless steel in ORR-MFE-4A, and 17.52 at. ppm H and 2.87 dpa in ORR-MFE-4B.</i></p>	
2.2	Operation of the ORR Spectral Tailoring Experiments ORR-MFE-4A and ORR-MFE-4B (Oak Ridge National Laboratory)	10
	<p style="margin-left: 40px;"><i>The ORR-MFE-4A experiment has operated for an equivalent of 391 d at 30-MW reactor power, with maximum specimen temperatures in the upper and lower regions of 400 and 330°C, respectively. A problem caused by a leak in the capsule primary system has been resolved, and the capsule is again in operation. The ORR-MFE-4B experiment, with specimen temperatures of 500 and 600°C, has operated for 283 d at 30-MW reactor power.</i></p>	
2.3	TEM Specimen Matrix for the AD-2 Reconstitution (Westinghouse Hanford Company)	13
	<p style="margin-left: 40px;"><i>TEM specimens have been prepared for inclusion in the reconstitution of the AD-2 test. The matrix includes specimens of HT-9 weld fusion zones and simulated heat-affected zones, undoped and nickel doped 12Cr-1Mo and 9Cr-1Mo alloys, and rapidly solidified Path A alloys.</i></p>	

2.4 HFIR/FFTF Irradiation Experiment – Conceptual Plan  
(Westinghouse Hanford Company and Oak Ridge National  
Laboratory) . . . . . 17

*Plans for a combined HFIR/FFTF irradiation experiment on Path A FCA and long-range ordered alloys are described. Irradiation will be up to -160 dpa with the appropriate helium levels expected for fusion environments. Irradiation temperatures will be from 400 to 600°C. A tentative matrix of swelling disks, creep tubes, and tensile, crack growth and fracture toughness specimens has been identified.*

2.5 Specimen Size Effects and Fusion Materials Research  
(Westinghouse Hanford Company) . . . . . 25

*Available data from specimens compatible with the Fusion Materials Irradiation Test (FMIT) facility have been compared with large specimen data to determine and quantify specimen size effects. For most of the properties examined, including properties where size effects are well documented, it was determined that the data obtained from small specimens are in good agreement with, and in some cases indistinguishable from, large specimen data. Small specimens however, require careful attention to the details of design, testing and analysis. Nevertheless, a good approximation of engineering or design quality data can be obtained despite the comparatively limited irradiation volume of fusion spectrum neutron radiation sources such as FMIT.*

2.6 Neutron Source Characterization for Materials Experiments  
(Argonne National Laboratory) . . . . . 66

*Data are presented from HFIR-CTR32, EBRII-X287, and the Omega West Reactor. An important new source of damage in nickel arises from the 340 keV <sup>56</sup>Fe recoil from the <sup>59</sup>Ni(n, α) reaction used to produce high helium levels in materials irradiations in a thermal spectrum. The status of all other experiments is summarized in Table 2.6.1.*

2.7 Elevated Temperature Irradiation of Ferritic Steel Charpy  
Specimens: Experiments HFIR-CTR-34 and -35 (Oak Ridge  
National Laboratory) . . . . . 87

*The HFIR-CTR-34 and -35 experiments contain miniature Charpy V-notch specimens of nickel-doped 12 Cr-1 MoVW and 9 Cr-1 MoVNb alloys. Different nickel contents result in different helium levels generated during irradiation in HFIR, to allow for an assessment of the effect of helium on impact properties. Irradiations at 300 and 400°C, to a midplane damage level of 10 dpa, have been completed.*

- 2.8 Elevated-Temperature Irradiation of Ferritic Steel Tensile Specimens: Experiments HFIR-CTR-39, -40, and -41 (Oak Ridge National Laboratory) . . . . . 91

*The HFIR-CTR-39, -40, and -41 experiments contain rod tensile specimens of the nickel-doped 9 Cr-1 MoVNb and 12 Cr-1 MoVW alloys and the 2 1/4 Cr-1 Mo alloy. Different nickel levels result in different helium levels generated during irradiation. Testing of these alloys will provide data on the effect of helium on tensile properties. Irradiations will be at 300, 400, and 500°C to a midplane fluence equivalent to 12 dpa.*

3. PATH A ALLOY DEVELOPMENT - AUSTENITIC STAINLESS STEELS . . . . . 97

- 3.1 Swelling of Path A PCA Irradiated to 10 dpa in HFIR (Oak Ridge National Laboratory) . . . . . 98

*The Path A PCA with various pretreatments and 20%-cold-worked type 316 stainless steel (CW 316) were irradiated in HFIR at 300 to 600°C to fluences producing up to 9.7 dpa and 620 at. ppm He. Very little melting was observed at 500°C and below, but swelling at 600°C ranged from 0.2 to 1.26. Swelling appears coupled to precipitate phase and dislocation microstructural development.*

- 3.2 Grain Boundary Microstructure in Path A PCA Irradiated to 10 dpa in HFIR (Oak Ridge National Laboratory) . . . . . 118

*Path A PCA in several preirradiation grain boundary microstructural conditions was irradiated in HFIR at 300 to 600°C to 10 dpa, together with 20%-cold-worked (N lot) type 316 stainless steel (CW 316). None of the samples show significant grain boundary bubble formation at 300 to 500°C. At 600°C, PCA-B1 and -B2, with stable, moderately sized grain boundary MC, have the finest dispersions of grain boundary bubbles.*

- 3.3 Fatigue Life at 650°C of 20%-Cold-Worked Type 316 Stainless Steel Irradiated in the HFIR at 550°C (Oak Ridge National Laboratory) . . . . . 136

*Strain controlled fatigue tests were performed in vacuum at 650°C on 20%-cold-worked type 316 stainless steel irradiated in the HFIR at 550°C. Fluences ranged from 0.85 to  $1.4 \times 10^{26}$  neutrons/m<sup>2</sup>. Little effect of irradiation was observed, but testing at 650°C failed to show the  $10^7$ -cycle endurance limit observed at 550°C.*

- 3.4 Irradiation Response of Rapidly Solidified Path A Type Prime Candidate Alloys (Massachusetts Institute of Technology) . . . . . 141

*The microstructural irradiation response of the Path A prime candidate alloy and two similar alloys with increased titanium and carbon content prepared by mpid solidification processing, has been investigated by neutron irradiation to 8.5 dpa (360 appm He) in the HFIR, and by dual ion irradiation to much higher dose (>100 dpa). The results show minor differences between conventionally prepared EA and EA prepared by mpid solidification techniques. However, these differences and the characteristic features of the response, the formation of very high densities of small cavities or bubbles and changes in the composition of initially existing TiC particles, are explained by a hypothesis of much importance to the mpid solidification approach. Our results suggest that increasing concentrations of titanium and carbon are beneficial, but only if the elements are initially in solution or present as very small (~2 nm) particles of TiC. The only way to practically achieve this for higher TiC contents than in PCA, is by using mpid solidification processing.*

- 3.5 Some Effects of Increased Helium Content on Void Formation and Solute Segregation in Neutron-Irradiated Type 316 Stainless Steel (Oak Ridge National Laboratory) . . . . . 169

*During EBR-11 irradiation of solution-annealed (SA) and 20%-cold-worked type 316 stainless steel (CW), void formation and melting are often coupled to radiation-induced solute segregation (RISS). A refined sink structure, including high cavity concentration, results if helium is preinjected before EBR-11 irradiation or produced during HFIR irradiation. This can suppress both the void swelling and the RISS.*

- 3.6 Application of Quantitative Electron Energy Loss Spectroscopy to Analysis of the Titanium Carbide Phase in Austenitic Stainless Steels (Oak Ridge National Laboratory and Argonne National Laboratory) . . . . . 217

*Quantitative analytical electron microscopy employing both x-ray energy dispersive spectroscopy (XEDS) and electron energy loss spectroscopy (EELS) was used to determine the carbon content of carbide particles in the Path A PCA. These results show that the experimental techniques must be adequate to resolve the Mo M and C K shell edges before quantitative measurement of carbon content is possible. Precipitate particles of different sizes had different metal-to-carbon ratios, which could be related to their titanium and molybdenum contents.*

- 3.7 Dynamic Powder Compaction of Rapidly Solidified Path A Alloy with Increased Carbon and Titanium Content (Massachusetts Institute of Technology) . . . . . 223

*Different techniques for consolidation of rapidly solidified alloys which are available or are under study at the present time include conventional consolidation techniques (hot extrusion, HIP, etc.), high velocity consolidation of atomized partially solidified particulates and dynamic powder compaction (DPC). This report describes the results of dynamic compaction of Path A alloy with increased carbon and titanium content. The microstructure of the as-compacted alloy is highly complex, evidencing an extreme degree of deformation. TEM revealed very high dislocation and twin density reflecting high hardness of the as-compacted alloy. Annealing studies revealed that recovery and recrystallization processes in dynamically compacted alloy are slower than in conventionally treated materials. High dislocation density appears to be an intrinsic property of the dynamic compaction process and it may be potentially useful in developing materials for irradiation performance. Other potential applications of dynamic compaction include preparation of graded materials and ceramic materials.*

- 3.8 Mechanical Properties and Structure of Y<sub>2</sub>O<sub>3</sub> Dispersion Stabilized, Rapidly Solidified 316 Type Stainless Steel (Massachusetts Institute of Technology) . . . . . 239

*An oxide dispersion stabilized 316 stainless steel has been prepared by mechanical alloying of a few 100 Å size yttria powders with attrited flakes of 316 stainless steel prealloyed with 1 wt% aluminum. Alloys with 4 vol% and 5 vol% yttria additions have been prepared and they show significant improvements in microstructural stability. Austenitic stainless steel containing 4 vol% yttria showed very high room temperature YS and UTS, with sufficient ductility. Stress rupture testing at 650°C as a function of different thermomechanical treatments is currently in progress.*

- 3.9 Microstructures Developed in Austenitic Stainless Steels Irradiated in HFIR at 55°C (Oak Ridge National Laboratory) . . . . . 250

*Samples of solution annealed types 316 and 347 stainless steel (SA 316 and SA 347) and 20%-cold-worked type 316 stainless steel (CW 316) have been irradiated in the High Flux Isotope Reactor (HFIR) at 55°C. No voids or bubbles were detected. All samples contained a fine "black spot" component of the dislocation structure. Both SA 316 and SA 347 contained a high concentration of Frank interstitial loops (14–30 nm diam) but no dislocation networks, whereas CW 316 shows network recovery with few Frank loops.*

3.10	Statistical Analysis of the MFE-5 In-Reactor Fatigue Crack Growth Experiment Results (Westinghouse Hanford Company) . . . . .	256
	<p><i>Results from the in-reactor fatigue crack propagation experiment and thermal control test have been quantitatively analyzed using statistical comparisons of the data. The analysis showed that at the 95% confidence limits, the in-reactor growth rates exceeds the thermal control by at most a factor of about two. However, the average growth rates in-reactor were less than that for the thermal controls, but statistically there were no significant differences between the crack growth rates from the two tests.</i></p>	
3.11	Tensile Properties of U.S.S.R. Austenitic Stainless Steel After Low-Temperature High Flux Isotope Reactor Irradiation (Oak Ridge National Laboratory) . . . . .	266
	<p><i>Two tensile specimens of the U.S.S.R. low-nickel, high-manganese stainless steel EP-838 were irradiated in the High Flux Isotope Reactor (HFIR) at about 55°C to produce about 5.2 dpa and 63 at. ppm He. The tensile properties at room temperature and 300°C of the irradiated and unirradiated 20%-cold-worked steel were quite similar to the properties of 20%-cold-worked type 316 stainless steel (CW 316) that was similarly irradiated.</i></p>	
4.	PATH B ALLOY DEVELOPMENT — HIGHER STRENGTH Fe-Ni-Cr ALLOYS . . .	277
	<p><i>No contributions.</i></p>	
5.	PATH C ALLOY DEVELOPMENT — REACTIVE AND REFRACTORY ALLOYS . . .	279
5.1	Mechanical Property Evaluations of Path C Vanadium Scoping Alloys (Westinghouse Electric Corporation) . . . . .	281
	<p><i>Nominal oxygen impurity levels of 600 and 1200 wppm, above the residual content of the alloys, have been introduced by gas-metal reaction into 0.76 mm sheet specimens of vanadium-base alloys. Tensile tests have been performed on each alloy at RT, 500, and 700°C and the results compared to previous tests on noncontaminated specimens. The oxygen contamination affected both strength and fracture behavior; these effects varied for the three alloys.</i></p>	
5.2	Swelling in Neutron Irradiated Titanium Alloys (Westinghouse Hanford Company) . . . . .	304
	<p><i>Immersion density measurements have been performed on a series of titanium alloys irradiated in EBR-II to a fluence of <math>5 \times 10^{22}</math> n/cm<sup>2</sup> (E &gt; 0.1 MeV) at 450 and 550°C. The materials irradiated were the near-alpha alloys Ti-6242s and Ti-56215, the alpha-beta alloy Ti-64 and the beta alloy Ti-38644. Swelling was observed in all alloys</i></p>	

with generally the greater swelling being observed at 550°C. Preliminary microstructural examinations revealed the presence of voids in all alloys. Ti-38644 was found to be the most radiation resistant. Ti-6242s and Ti-5621S also displayed good radiation resistance, whereas considerable swelling was observed in Ti-64 at 550°C.

6. INNOVATIVE MATERIAL CONCEPTS . . . . . 313

6.1 Microstructures of Iron-Base Long-Range-Ordered Alloys Bombarded to 70 dpa with Nickel Ions (Oak Ridge National Laboratory) . . . . . 314

*Disks of LRO-37-5 in the as-ordered and 20%-cold-worked conditions were bombarded with 4-MeV nickel ions to 70 dpa at temperatures from 570 to 680°C. Both conditions retained order for bombardment below the critical ordering temperature of about 650°C, and cold work improved the swelling resistance, especially above  $T_c$ . Both conditions showed better resistance to swelling than 20%-cold-worked type 316 stainless steel.*

6.2 Status of Scale-Up of an Iron-Base Long-Range-Ordered Alloy (Oak Ridge National Laboratory) . . . . . 321

*The semiproduction scale-up of an iron-base LRO alloy is under way by a commercial source. Three 18-kg ingots have been produced by two different melt practices. Chemical analyses indicate that the compositions of the ingots are near the alloy specifications. Specimens from the ingots were successfully cold rolled at room temperature and hot rolled at 1100°C. Further evaluation of alloy homogeneity and distribution of second phases is in progress.*

6.3 Microstructures of Iron-Base Long-Range-Ordered Alloys Irradiated to 10 dpa in HFIR (Oak Ridge National Laboratory) . . . . . 327

*The LRO-16, -20, and -37 alloys were irradiated in HFIR at 300 to 600°C to nominally 8.3 dpa and 1000 at. ppm He. All three remained ordered, and interstitial loops, dislocation segments, and cavities formed in the microstructure. Cavities grew preferentially on {111} loops, and helium bubbles formed on grain boundaries at the higher temperatures. Swelling in LRO-20 and -37 was low; LRO-16 swelled more than annealed EA.*

7. PATH E ALLOY DEVELOPMENT — FERRITIC STEELS . . . . . 335

7.1 Postirradiation Notch Ductility and Fracture Toughness Behavior of AOD Heat of Alloy HT-9 (Naval Research Laboratory) . . . . . 336

*Alloy HT-9 is being evaluated for potential application as a first wall material in magnetic fusion reactors. One objective of the current studies is the assessment of material notch ductility and fracture toughness in the pre- and postirradiation conditions.*

*Charpy-V ( $C_V$ ) and fatigue precracked Charpy-V (PCC<sub>V</sub>) specimens of a 1.7 cm thick plate from the HT-9 reference melt (AOD process) were irradiated at 93° and at 288° C to  $\sim 8 \times 10^{19}$  n/m<sup>2</sup>,  $E > 0.1$  MeV, in a water-cooled test reactor. The  $C_V$  transition temperature elevation produced by the 93° C irradiation was more than three times that produced by the 288° C irradiation. The 93° C irradiation data showed the alloy to be unacceptable for 93° C service unless (only) elastic fracture resistance is acceptable. Good agreement was observed between  $C_V$  and PCC<sub>V</sub> determinations of radiation-induced embrittlement.*

7.2 Impact Test Results for Irradiated Ferritic Alloys (Westinghouse Hanford Company) . . . . . 342

*To be reported in the next semiannual report.*

7.3 Microstructural Examination of HT-9 and 9Cr-1Mo Contained in the AD-2 Experiment (Westinghouse Hanford Company) . . . 343

*The microstructures of HT-9 and modified 9Cr-1Mo have been examined before and after irradiation in EBR-II in the AD-2 experiment to a fluence of  $2.5 \times 10^{22}$  n/cm<sup>2</sup> ( $E > 0.1$  MeV) at temperatures from 400 to 550° C. The precipitate structure of unirradiated HT-9 consists of  $M_{23}C_6$  which forms at martensite lath and prior austenite grain boundaries. 9Cr-1Mo shows much less  $M_{23}C_6$  at grain boundaries, but contains finely-dispersed (Nb, V, Cr)C within grains. Irradiation of both alloys at 400° C causes extensive additional precipitation, formation of a dislocation substructure, and, in 9Cr-1Mo, void formation. The irradiation-induced precipitate in HT-9 was identified as G-phase, a nickel silicide with fcc crystal structure and lattice parameter of 1.12 nm. Only very small additions of nickel are required in ferritics to promote the formation of G-phase. 9Cr-1Mo irradiated at 400° C formed rod-shaped Cr<sub>2</sub>C. Above 450° C, the HT-9 formed additional  $M_{23}C_6$ , whereas the 9Cr-1Mo formed additional MC and a phosphide with Fe, Cr, and Mo. No voids were observed above 450° C in the 9Cr-1Mo alloy. The formation of finely-dispersed G phase in HT-9 and Cr<sub>2</sub>C in 9Cr-1Mo during irradiation below 450° C is believed to contribute to observed irradiation hardening and increased DBTT.*

7.4 The Procurement and Characterization of the Electroslag Remelted National Fusion Program Heat of 12Cr-1Mo Steel (General Atomic Company) . . . . . 363

*Previously it was reported that a 30,000 lb heat of a 12Cr-1Mo steel melted to the specification of Sandvik HT-9 was procured by General Atomic Company from Electralloy Corporation for the Ferritic Steel Program. This heat of argon-oxygen decarburization (AOD) 12Cr-1Mo steel has, since, been electroslag remelted (ESR) by Universal Cyclops Corporation in Pittsburg, and converted to plate and bar stock. This remelted heat (No.9607R2) was within the chemical specification requirement of HT-9. The remelted ingot was homogenized at 1100C for 16 hrs prior to conversion to the product forms. After conversion to plate and bar stock, each form was given a stress relief at 700C for 6 hrs. Subsequent mechanical property analysis on the 0.625 in (1.58 cm) plate has shown that it meets the strength and tensile elongation specifications. The UTS was 121 ksi (835 MPa), the yield stress was 87.5 ksi (604 MPa), and the elongation was 21%. The upper shelf energy determined on standard Charpy impact specimens was 85 ft-lbs (115J) as compared to the upper shelf of the AOD heat of 66 ft-lbs (89J). Microstructural examination showed a uniform tempered martensitic structure with a delta ferrite content less than 1%.*

7.5 The Weldability of HT-9: Preheat and Position Effects (General Atomic Company) . . . . . 370

*To be reported in the next semiannual report.*

7.6 The Toughness of Simulated Heat-Affected Zone Microstructures in a 12Cr-1Mo-0.3V Martensitic Stainless Steel (Sandia National Laboratories) . . . . . 371

*The four distinct HAZ microstructures which were identified in HT9 GTA welds were simulated in the Gleeble in order to produce bulk microstructural Charpy V-notch samples suitable for evaluation of notch toughness. Samples both transverse and longitudinal to the rolling direction (R.D.) of AOD melt plate stock were evaluated. Following the Gleeble thermal treatment the samples were given a postweld heat treatment (PWHT) for 1 hour at 760° C. Charpy tests of the individual microstructures were performed over the temperature range from -60 to 200° C. The toughness values of samples oriented transverse to the rolling direction (notch aligned parallel to R.D.) fell in a wide scatterband which enclosed the base metal toughness at low temperatures and was slightly less than the base metal value on the upper shelf. When samples oriented parallel to the R.D. (notch transverse to R.D.) were evaluated the toughness of three of the microstructural regions was again equivalent to that of the base metal.*

*The fourth region, which contained a fine prior austenite grain size exhibited superior toughness properties with respect to both the ductile-to-brittle transition temperature (DBTT) and the upper shelf energy.*

**7.7 The Effect of Postweld Heat Treatment on the Toughness of the Heat-Affected Zone in a 12Cr-1Mo-0.3V Steel (HT9) (Sandia National Laboratories) . . . . . 388**

*Samples both transverse and longitudinal to the rolling direction (R.D.) of the plate were subjected to thermal cycles in the Gleeble which simulated four distinct microstructures in the HAZ. Following the Gleeble heat treatment, the samples were tempered at either 760°C or 600°C for one hour and subsequently tested over a range of temperatures in the Charpy apparatus. Charpy test results revealed that the higher PWHF decreased the ductile-to-brittle transition temperature (DBTT) and increased the upper shelf energy of three of the microstructures. A fourth microstructure was relatively unaffected by the PWHF since the peak temperature experienced by this HAZ region is not sufficient to reaustenitize the structure during welding. The region of the microstructure heated just slightly above the austenitization temperature during welding exhibits the best toughness properties due to the refinement of the austenite grain size.*

*The Charpy results indicate that a 600°C postweld heat treatment is not sufficient to increase the toughness of the weld region to a level which would guarantee the structural integrity of a large component. It is imperative that the ambient temperature toughness should be sufficient to ensure extended life during cyclic loading and periodic shutdown of the reactor.*

**7.8 Interpretive Report on the Weldability of 12Cr-1Mo-0.3V (HT9) Martensitic Stainless Steel, Part 2: Suitability for Fusion Reactor First Walls and Blanket Structures (Sandia National Laboratories) . . . . . 399**

*A previous report summarized the results of both commercial experience and ADP-sponsored research concerned with the weldability of the HT9 candidate alloy. The general conclusion of this report was that HT9 exhibits adequate weldability and is suitable from a fabrication standpoint; however, the question of in-service behavior and survivability in an operating reactor have not been addressed. To date, the data to make such judgments are not available. Continuing research in the areas of irradiation damage, fatigue behavior, hydrogen embrittlement, and the synergism among these issues and others must be pursued before the suitability of HT9 weld microstructures in fusion reactor environments can be properly assessed.*

7.9 Hydrogen Embrittlement of ESR Processed 12Cr-1Mo Steel (Sandia National Laboratories) . . . . . 401

*Tensile tests were performed to determine the effect of hydrogen on the ductility of a quenched-and-tempered 12Cr-1Mo alloy from the National Fusion Heat-ESR processed. specimens were cathodically charged with hydrogen at 0.003 A/cm<sup>2</sup> and 0.006 A/cm<sup>2</sup> for 15 to 150 minutes. In the uncharged condition, the ESR specimens failed by intergranular rupture with prominent secondary cracking along prior austenite grain boundaries. For the least severe cathodic charge (0.003 A/cm<sup>2</sup> - 15 minutes), the tensile ductility (R<sub>A</sub>) decreased 33% and the degree of secondary cracking increased. At 0.003 A/cm<sup>2</sup> for 150 minutes, the ductility decreased further and the fracture mode changed to classical intergranular fracture. These results generally matched those for the AOD processed 12Cr-1Mo alloy that had been tested previously.*

*Auger analysis indicates that there is segregation of phosphorous and the carbide-forming elements Cr, Mo, W to the prior austenite grain boundaries. In the literature, phosphorous has been shown to embrittle these boundaries, especially in the presence of hydrogen.*

*The combination of tensile data on hydrogen charged ESR and AOD 12Cr-1Mo steel present several reasons for concern regarding the susceptibility of this alloy to hydrogen embrittlement. Earlier results of hydrogen charged Sandvik HT-9 are addressed in terms of these more recent findings.*

7.10 Tensile Properties of Ferritic (Martensitic) Steels After Low-Temperature HFIR Irradiation (Oak Ridge National Laboratory) . . . . . 414

*In addition to the ferritic steels previously tested, experiment HFIR-CTR-33 also contained tensile specimens of 20%-cold-worked type 316 stainless steel for irradiation at about 55°C. The tensile behavior of the type 316 stainless steel was determined at room temperature and 300°C, and the results showed the irradiated properties of the two classes of steels to be quite similar.*

7.11 Helium Embrittlement Tests on Ferritic Steels (Oak Ridge National Laboratory) . . . . . 424

*Tensile tests at 700°C a nickel-doped 9 Cr-1 MoVNb and 12 Cr-1 MoVW steels that were irradiated at 55°C in HFIR to produce up to about 50 at. ppm He indicated that there was no helium embrittlement. These results contrast with literature data showing that very small amounts of helium can cause a large decrease in ductility for many alloys when tested under equivalent conditions.*

**7.12 Reconstitution of the AD-2 Ferritics Experiment  
(Westinghouse Hanford Company) . . . . . 431**

*Six uninstrumented B-7c capsules were removed from the EBR-II after undergoing irradiation during Cycles 109-113. Several specimens were removed and distributed for interim examination, while the remaining specimens were re-encapsulated into four new B-7c capsules. In addition, some irradiated specimens were included in the new capsules. Irradiation continued with Cycle 118 and will end after Cycle 123.*

**7.13 Fractographic Examination of Compact Tension Specimens of Unirradiated HT-9 and Modified 9Cr-1Mo Welds (Westinghouse Hanford Company) . . . . . 442**

*Miniature compact tension specimens of HT-9 and modified 9Cr-1Mo weld metal and HT-9 HAZ material have been tested and examined by scanning electron microscopy in order to provide baseline data for comparison with irradiated specimens, to provide understanding of the fracture process in these materials, and to assess the usefulness of crack opening displacement measurements based on fractographic analysis. For specimens tested at 205°C, crack propagation is found to be due to microvoid coalescence.*

**7.14 Postirradiation Notch Ductility of the Weld Heat Affected Zone (HAZ) of Alloy HT-9 Plate (AOD Heat) (Naval Research Laboratory and Sandia National Laboratories) . . . . . 460**

*Alloy HT-9 is being evaluated for potential application as a first wall material in magnetic fusion reactors. The size and complexity of projected components may necessitate the use of welding for component fabrication; accordingly, studies of Alloy HT-9 irradiation resistance capabilities are including assessments of weld deposits and weld HAZ materials. The present investigation represents a joint effort by the Naval Research Laboratory (NRL) and the Sandia National Laboratories at Livermore (SNLL).*

*Charpy-V ( $C_v$ ) specimens simulating four positions across a weld HAZ and specimens of the parent plate material were irradiated at 288°C to  $-8 \times 10^{19}$  n/cm<sup>2</sup>,  $E > 0.1$  MeV. The HAZ specimens were thermally cycled on a Gleeble apparatus; individual thermal cycles provided peak temperatures of 1380, 1152, 974 or 828°C. Postirradiation notch ductility determinations revealed the HAZ to have essentially uniform irradiation embrittlement sensitivity across its width. In addition, the data showed the irradiation resistance level of the HAZ to be comparable to that of the parent plate. The results in turn suggest that, for the 288°C service condition, the fracture resistance of irradiated HT-9 weldments will not be governed primarily by the weld HAZ.*

8. STATUS OF IRRADIATION EXPERIMENTS AND MATERIALS INVENTORY . . . 467

8.1 Irradiation Experiment Status and Schedule (Oak Ridge National Laboratory) . . . . . 46%

*principal features of many ADIP irradiation experiments are tabulated. Bar charts show the schedule for recent, current, and planned experiments. Experiments are presently under way in the Oak Ridge Research Reactor (ORR) and the High Flux Isotope Reactor (HFIR), which are mixed spectrum reactors, and in the Experimental Breeder Reactor (EBR-II), which is a fast reactor.*

8.2 ETM Research Materials Inventory (Oak Ridge National Laboratory) . . . . . 476

*The office of Fusion Energy has assigned program responsibility to ORNL for the establishment and operation of a central inventory of research materials to be used in the Fusion Reactor Materials research and development programs. The objective is to provide a common supply of material for the Fusion Reactor Materials Program. This will minimize unintended materials variables and provide for economy in procurement and for centralized record-keeping. Initially this inventory will focus on materials related to first-wall and structural applications and related research, but various special purpose materials may be added in the future.*

9. MATERIALS COMPATIBILITY AND HYDROGEN PERMEATION STUDIES . . . . 481

9.1 Corrosion of Austenitic, Ferritic, and Long-Range-Ordered Alloys in Flowing Lithium (Oak Ridge National Laboratory) . . . . . 482

*Results from lithium thermal-convection loop (TCL) tests of type 316 stainless steel, Sandvik HT9, and a long-range-ordered (LRO) alloy composed of Fe-31.8 Ni-22.5 V-0.4 Ti (wt %) are reported. Type 316 stainless steel and HT9 had similar steady-state dissolution rates at 500°C. The LRO alloy was rapidly corroded in type 316 stainless steel TCLs. However, results from isothermal tests in lithium showed that dissimilar-metal transfer probably made a very significant contribution to the overall LRO alloy corrosion rate measured in the above experiments.*

9.2 Environmental Effects on Properties of Structural Alloys (Argonne National Laboratory) . . . . . 491

*Compatibility tests were conducted with several candidate structural materials to study the corrosion behavior in flowing lithium, and fatigue tests were performed with HT-9 alloy specimens that were preexposed to lithium. The results indicate that the corrosion rate of ferritic steels*

is an order of magnitude lower than for the austenitic stainless steel. The corrosion rate for cold-worked Type 316 stainless steel is a factor of 3 greater than that for the annealed steel. Preexposure (1000 h) of the HT-9 alloy to low-nitrogen lithium has no effect on fatigue life. Preliminary scoping studies indicate that V-15Cr and V-15Cr-5Ti alloys are highly corrosion resistant in 523 K (250°C) deoxygenated water.

**9.3 Corrosion of Austenitic and Ferritic Steels in Static Pb-17 at. % Li (Oak Ridge National Laboratory) . . . . . 500**

Specimens of type 316 stainless steel and Sandvik HT9 were exposed to static, molten Pb-17 at. % Li. Weight losses of these alloys exposed to Pb-17 at. % Li were much greater than those measured in static, pure lithium. Decreased weight losses at longer exposure times were thought to be a result of possible corrosion product formation on the specimen surfaces due to impurities in the melt.

**9.4 Compatibility Studies of Structural Alloys with Solid Breeder Materials (Argonne National Laboratory) . . . . . 507**

The compatibility of solid  $\text{Li}_2\text{O}$ ,  $\text{LiAlO}_2$ , and  $\text{Li}_2\text{SiO}_3$  breeder materials with Type 316 stainless steel and ET-9 alloy has been investigated at 773 K (500°C). The results show that for  $\text{Li}_2\text{O}$ , the alloy-ceramic interactions at 773 K are similar to that observed at 973 K (700°C). Both steels show a uniform layer of internal penetration and a thick outer scale that consists of the ceramic material embedded with iron-rich corrosion products. Specimens exposed with  $\text{LiAlO}_2$  or  $\text{Li}_2\text{SiO}_3$  show no measurable interaction. A compatibility test with  $\text{Li}_2\text{O}$  has been initiated at 823 K (550°C) in a flowing helium environment containing controlled amounts of moisture and hydrogen.