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Alloy Development for Irradiation Performance

Quarterly Progress Report

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Assistant Secretary for Energy Technology

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The Materials and Radiation Effects branch of the Office of Fusion Energy has established a mterials data book for use in fusion related studies. This handbook will be known as the Materials Handbook for Fusion Energy Systems and initial copies should be available in the near future. The handbook will be part of the Analysis and Evaluation subtask group of ADIP and will establish the data requirements and priorities for the preparation of the individual data sheets. At the present time Analysis and Evaluation suhtask group has given the data sheets to support the ETF study the first priority. Based on recom mendations from R. E. Nygren (ETF-Materials) the first data sheets will be those relating to the austenitic stainless steels.

1.2 Assessment of Cyclic Creep-Fatigue Life of First-Wall Structural Material (Oak Ridge National Laboratory) 5

Cyclic lifetimes for a fusion reactor first wall were independently evaluated with two methods: linear damage summation (LDS), which is currently being employed for use in ASME high-temperature design Code Case N-47, and strain-range partitioning (SRP), which is being studied for inclusion in the design Code. Despite their diversity, results estimated with the two analytical techniques agreed reasonably well in several practical design situations. Furthermore, this excercise has reaffirmed our earlier finding that 20%-cold-worked type 316 stainless steel could be used as a fusion first-wall material meeting a 8 to 10 MW-year/m^2 lifetime goal for a neutron wall loading of about 2 MW/m^2 and a maximum temperature of about 500°C.

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Fracture toughness tests have been performed on miniature circular tension specimens of HT-9 at room temperature. The shape and size of specimens were tailored for efficient stacking in EBR-II irradiation. Furthermore, the number of specimens required will be reduced by using a single specimen method. The electropotential technique has been applied to obtain a calibration curve for voltage change versus crack extension. Using this calibration curve, J-integral measurements are imde. They are in agreement with those from a multi-specimen method.

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5.2 Microstructure of Titanium Alloys After EBR-II Irradiation (McDonnell Douglas Astronautics Company — St. Louis and McDonnell Douglas Research Labs) 42

Bulk annealing experiments performed on Ti-6Al-4V indicate that the beta precipitate observed, following irradiation, was dissolved after annealing for 24 hours at 650°C. The temperature where the precipitate dissolved was well below the beta transus and appears to support the contention that the precipitate is irradiation induced and not a result of supersaturation of the alpha phase. Instead the precipitate appears to be caused by the alpha solute segregation at vacancy sinks. These sinks appear to act as traps for the vanadium atoms which have a high diffusion mobility as titanium because of its smaller atom size. The presence of a larger concentration of vanadium at the defect site results in alteration of the equilibrium concentration resulting in the precipitation of the beta phase. The near alpha alloys Ti-6242S and Ti-56215 use molybdenum as a beta stabilizer, however, because of its larger atom size and low concentration, it effectively suppresses the irradiation beta formation under these irradiation conditions.

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The planning of this study is complete. Modified wedge opening loading (modified WOL) specimens are being machined from 0.32 cm (0.125 inch) thick Ti-6242s. One half of the specimen blanks are being charged with hydrogen to a desired bulk level of 500 wppm with the hydrogen uniformly distributed throughout the samples.

5.4 Hydrogen in Titanium Alloys (McDonnell Douglas Astronautics Company — St. Louis) 53

A review of current knowledge concerning the effects of hydrogen on the mechanical properties of titanium has provided minimal data directly applicable to fusion reactor usage. Although no definite conclusions have been drawn as yet, it presently appears that because of the large number of interdependent variables associated with hydrogen interactions it may be impractical to predict with any degree of certainty the embrittlement effects, if any, for the alloys of interest for fusion energy. There are some indications that titanium alloys m y not be adversely degraded in the elevated temperature, low hydrogen pressure conditions in the reactor.

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The AD-1 experiment consists of three uninstrumented pins containing microstructural and mechanical property specimens made from titanium alloys. These three pins were designed for irradiation temperatures of 394°C, 450°C and 500°C and have been irradiated in EBR-11 until fluences greater than $4 \times 10^{22} n/cm^2$ (E>0.11 MeV) have been attained. Evidence of corrosion was observed on several specimens contained in the subcapsule designed for an irradiation temperature of 500°C. The subsequent opening of the other two subcapsules has been delayed until more information has been obtained concerning this corrosion.

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mechanical behavior. Their strength, instead of decreasing as with conventional alloys, increases with temperatures because of ordering. The ordered alloys also exhibit excellent creep strength. Their creep rate at 650°C is lower than that of type 316 stainless steel by more than three orders of magnitude.

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Small heats of HT9 and 9 Cr-1 Mo steels have been prepared containing 1 and 2% Ni. An additional heat of each alloy was prepared with 2% Ni but with adjustments in the carbide-forming elements to restore the chromium equivalent to a value near that of the original alloy. Tensile and transmission electron microscope (TEM) specimens of these mterials will be irradiated in the High Flux Isotope Reactor (HFIR). The thermal flux in the HFIR will result in the formation of helium in concentrations approximating those produced in fusion reactor service.

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The AD-2 experiment will consist of six, uninstrumented B-7c pins containing microstructure and mechanical property specimens made from ferritic alloys. Alloy selection, irradiation temperatures and the test matrix have been specified and approved by the Alloy Development for Near Term Applications Working Group. Approval in principle has been granted by DOE for the AD-2 experiment and irradiation will begin at EBR-11 in June, 1980.

7.4 **Environment Effects on Properties of Ferritic Steels (Argonne National Laboratory)** 115

This program was initiated in the first quarter of FY 1980 with initial phases of the program focused on (1) combined effects of stress and coolant environment on the properties of ferritic steels and (2) the compatibility of ferritic steels with candidate solid tritium-breeding materials and neutron multipliers. The first experiments include determination of the effects of a liquid lithium environment on the fatigue properties of the Sandvik RT-9 alloy and tests on the compatibility of candidate ferritic steels with Li_2O and Li_2SiO_3.

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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 materials for the Fusion Reactor Materials Program. This wilt minimize unintended materials variables and provide for economy in procurement and for centralized record-keeping. Initially this inventory is to focus on materials related to first-wall and structural applications and related research, but various special purpose materials may be added in the future.

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Efforts to characterize the hydrogen dissolution and permeation characteristics of alloys currently under development in the ADIP Program have continued. An effort to measure the hydrogen permeability of pure α-titanium was generally unsuccessful owing to a tendency of the material to creep in the gasket-seal-type assembly normally used for difficult-to-weld alloys. Work a the solubility of hydrogen in Ti-6Al-4V was completed. Rased on these results, the amount of hydrogen dissolved in a Ti-6Al-4V fusion reactor first wall assembly could range from several

hundred grams to a few kilograms, depending on the effective average T_2 pressure in the plasma boundary region and the first wall temperature distribution. Results of a survey of the hydrogen permeation characteristics of ferritic alloys showed that they are ~5-times more permeable than 300-series stainless steels in the surface-unperturbed state, but that significant reductions in permeability can be achieved for oxidized aluminum-and chromium-containing ferritics.

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The stainless steel-clad vanadium alloy loop with circulating lithium has continued to operate through the first quarter of FY 1980. Test samples of zirconium, vanadium, V-15Cr-5Ti, titanium, and yttrium have been exposed at 873 K to investigate the distribution of nonmetallic elements in lithium/refractory metal systems. The zirconium and yttrium exposures have been used to adjust and evaluate the nonmetallic (O, N, H, and C) concentration in lithium. Results of hydrogen analyses of the test samples have been compared with calculated distribution coefficients for hydrogen. Similar analyses of oxygen and nitrogen distribution are in progress.

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Data are reported on the compatibility of 500 and 600°C lithium with alloy 800 [46 Fe—32 Ni—21 Cr (wt %)] and a long-range-ordered (LRO) alloy [45 Fe—32 Ni—23 V (wt %)]. Although these alloys contained similar concentrations of Fe and Ni, alloy 800 was severely corroded in certain instances, while the LRO alloy resisted attack and incurred no loss of tensile properties in Li.

9.4 Corrosion of Type 316 Stainless Steel and Alloys B1 and B3 in Lithium Thermal-Convection Loops (Oak Ridge National Laboratory) 158

Results from three thermal-convection-loop experiments are reported. The corrosion rates of ADIP alloys B1 and R3 [64 Fe—25 Ni—10 Cr—1 Mo (wt %) and 56 Fe—30 Ni—12 Cr—3 Mo (wt %)], respectively1 were much greater than that of type 316 stainless steel. Similar corrosion mtes in flowing lithium were measured for type 316 stainless steel in both austenite and a-ferrite systems. Adding aluminum to a type 316 stainless steel loop that had already been circulating lithium for over 5000 h resulted in flow restrictions that were probably the result of aluminum deposits in the coldest part of the circuit. Consequently, corrosion inhibition by aluminum addition to the lithium is most effective in a new system.

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