File Copy
Do not remove

DOE/ER-0045/4

Alloy Development for Irradiation Performance

Quarterly Progress Report
For Period Ending September 30, 1980

U.S. Department of Energy
Office of Fusion Energy

CONTENTS

In mid-September a meeting was held by the austenitic stainless steel working group of ADIP. The purpose of this meeting was to review the prepared data sheets on 20% CW type 316 stainless steel. The consensus of this group was that some minor modifications needed to be made prior to their release. The first of these data sheets which covers the effect of irradiation on the elevated temperature fatigue strength of 20% cold worked 316 stainless steel has been received by the MHFES and subsequently submitted to the Analysis and Evaluation task group for review and approval.

The cross sections that include the delayed fission gammas have been processed. Additional neutronic calculations using these cross sections have been carried out to determine the tritium production and heating rates within two fusion breeder mterials, Li_2O and $LiAlO_2$. During the generation of the new cross-section set we detected an error in the thermal ^{235}U neutron cross section. Correcting this error resulted in an increase in the tritium production and heating rates within the test assembly of about 30 and 10%, respectively, for the Li_2O breeder mterial.

Since the temperature gradient across the test cell should be as small as possible, we have changed the original geometry. The original outer dimensions of the test cell have been reduced from 17.8 to 12.7 mm and the stainless steel can from 19.1 to 14.0 mm. The aluminum core piece inner radius was reduced to 14.0 mm to fill the void. Additional neutronic calculations using the above geometrg changes have been carried out to determine the changes in the tritium production and heating rates. The rates for the new geometric model are similar to those obtained for the original model. Preliminary burnup calculations of the Li_2O breeder material indicate that one-half the ^6Li will be consumed in ahout 70 d reactor operation.

Three-dimensional neutronic calculations are being carried out to follow the irradiation environment of the ORR-MFE-4A eqeriment. These calculations currently cover six ORR reactor cycles corresponding to 46,742 MW hrs. This will result in a thermal fluence of 2.03×10^{25} neutrons/m^2 and a total fluence of 4.31×10^{25} neutrons/m^2. This fluence will produce 1.09 dpa and 3.31 at. ppm He in type 316 stainless steel (not including 2.0 at. ppm He from ^{10}B). These data and previous calculations have been used to estimate the dates at which the core pieces should be changed and first samples removed.

Through preliminary calculations we have determined the appropriate sizes of the tungsten core pieces as well as the heating mtes within them and within the experimental capsules. The reduction in gamma heating resulting from the use of tungsten core pieces may require additional heating of the samples to maintain proper temperature. Calculations similar to those carried out for the tungsten core pieces art? in progress for hafnium core pieces.

This eqeriment consists of two test regions designed to irradiate type 316 stainless steel and Path A PCA at constant temperatures of 300 and 400°C. The ORR-MFE-4A experiment was installed in the Oak Ridge Research Reactor (ORR) on June 10, 1980, and as of September 30 has operated successfully for an equivalent 99 d at 30 MW reactor power with maximum specimen temperatures in each region of 330 and 400°C respectively.

Three eqeriments for the irradiation of ferritic steels in the HFIR have been outlined. All three experiments are to operate at about 50°C. They wilt provide the first substantial data on the effects of helium and displacement production on several properties, including fatigue, tensile, Charpy impact, crack growth, and fracture toughness. The steels to be tested include 12 Cr-1 Mo (HT9), nickel-doped 12 Cr-1 Mo, 9 Cr-1 Mo, nickel-doped 9 Cr-1 Mo, 2 1/4 Cr-1 Mo, and welds of some of these. Test results will complement elevated-temperature Experimental Breeder Reactor (EBR-II) irradiation experiments on the same alloys. Test nntrices and fluence levels have been defined, and the eqeriments are expected to be ready for insertion in late 1980 or early 1981.

Laser welds were made in 6.35mm (0.25 in) HT9 plate using a variety of tmvel speed/focal length combinations at a constant laser power level of 6kW. Velds performed at sharp focus with weld travel speeds ranging from 1.27 to 4.23 mm/sec exhibited scattered porosity and occasional centerline cmcking. Defocusing the laser beam relative to the plate resulted in welds containing severe porosity and centerline cracks. Microstructural and microhardness evaluation of the welds indicated that the region of highest hardness occurred in the heat-affected zone immediately adjacent to the fusion line. A mechanism for centerline cracking in this alloy has been proposed.

Several continuous-cycle fatigue tests have been conducted with 2.5-m-diameter specimens of HT-9 alloy at 755 K in flowing lithium. The results indicate a strong effect of corrosion on the fatigue life of HT-9 alloy in a liquid lithium environment. At a total strain mnge of 0.5%, the fatigue life in lithium is a factor of -5 lower than that in a liquid sodium environment. The specimens tested in lithium show intergranular cracks along the entire gauge length. Fatigue tests at different strain rates and strain sequence are being conducted to establish the important parameters, viz., stress/strain range, frequency, lithium purity, etc. Exposure of corrosion specimens of HT-9 alloy, Type 316 stainless steel, and Inconel 625 with solid Li_2O, $LiAlO_2$, and Li_2SiO_3 breeding materials at 873 K has been completed. Metallographic evaluation of the specimens is in progress. Preliminary results indicate that lithium oxide is the most reactive of the three breeding materials.

Calculations of the hydrogen level in first wall and blanket structures have been made using HT9 as the material of construction. Both directly injected deuterium and tritium profiles cmd hydrogen profiles from (n,p) reactions have been calculated over a temperature mnge from 473 to 638K. Two boundary conditions have been assumed: zero surface concentration and a concentration set by the surface recombination reaction of hydrogen ions to form hydrogen molecules. The results indicate that under the most severe conditions, peak hydrogen levels will not exceed 0.5 appm and wilt likely be far lower.

The following bar charts show the schedule for all 4DIP reactor irradiation 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.

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.

Studies of the hydrogen permeation characteristics of iron base (Path E) alloys have continued. An investigation of a Cr 14-A1 0.2 steel (405-SS) was completed and the measurements were compared to previous data for three other Fe-Cr alloys having 0, 2, and 5 wt.% Al, respectively. The combined results indicate that the optimum aluminum content leading to minimum hydrogen permeability in a reducing environment is around 2 wt.%. However, the chromium content of alloys studied to date has only spanned the range from 14 to 18 wt.%, and the location of an optimum at 2 wt.% aluminum may not necessarily apply to other types of Path E alloys.

The stainless-steel-clad/V-15 Cr lithium loop has now operated for over 12,000 h and exposures of selected refractory alloys have continued. The DOE/Office of Fusion Energy guidance on this project for FY-1981 has been to defer effort to solid breeder mterial development activities. The work presently in progress is being completed and a final reporting of results will be made in a subsequent ADIP progress report.

Specimens of 2 1/4 Cr-1 Mo steel, HT9, and the long-range-ordered (LRO) alloy Fe–31.8 Ni–22.5 V–0.4 Ti (wt %) were tested in static lithium. Carbon analysis of the lithium after 3000-h tests of 2 1/4 Cr-1 Mo steel showed significant decarburization of the steel at 500 and 600°C. The decarburization lowered the room temperature tensile strength of the alloy. Specimens of 2 1/4 Cr-1 Mo steel exposed to Li–5 wt % Al showed significant weight gains with an accompanying decrease in ductility. Also, the gage section of fractured tensile specimens contained a significant volume of cracks. Short-term (500-h) exposures of HT9 to static 500°C lithium resulted in negligible weight changes and no change in its tensile properties relative to specimens exposed to argon under otherwise similar conditions. Surface deposits on the corrosion-resistant LRO alloy after exposure to lithium at 650 and 710°C for 2000 h were pure vanadium or possibly a vanadium carbonitride.

The time dependence of metal dissolution in five type 316 stainless steel loop experiments in lithium is discussed. In general, the five sets of measurements were satisfactorily reproducible. The predicted dissolution rate of type 316 stainless steel at 600°C in lithium under conditions typical of semistagnant tritium-breeding blankets is less than 7% μm/year (0.5 mil/year).

An initial scoping experiment was carried out to investigate the interfacial compatibility of selected solid ceramic tritium breeder materials with typical ADIP Program alloy.?. Each of the solid breeder materials Li_2O, $LiAlO_2$, and Li_2SiO_3 was exposed to 316-SS, HT-9, Inconel 625, and Ti6242 at 873 K for ~1900 h in a high purity helium environment using a reaction couple method. Examination of the alloy/ceramic interfaces by SEM, Auger, and X-ray diffraction analysis revealed that reaction scales comprised of elements from both the alloy and ceramic had formed in all cases. These scales were thickest for the Li_2O/alloy reaction couples. Ternary phases of the type $Li_xM_yO_z$ have been identified at most of the interfaces for which analyses have been completed. Analytical procedures and improved strategies for future ceramic breeder corrosion tests have been developed.