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boundary chemistry measurements. The microstructural, microchemical, and Charpy impact response of V-4Cr-4Ti displayed trends similar to those observed for V-5Cr-5Ti. The results show that grain size plays an important role in determining the ductile-to-brittle transition temperature (DBTT) of these materials and that a threshold level of grain boundary segregant appears to be required to cause grain boundary embrittlement and intergranular fracture.

Extended Abstract

Extended Abstract

Mechanical properties of V-4Cr-4Ti alloy were investigated after low-temperature (<420°C) irradiation. The effects of fast neutrons at 390°C were investigated by irradiation to ≈4 dpa in the X530 experiment in the EBR-II reactor; these tests were complemented by irradiation with the single (4.5-MeV Ni^{++}) and dual ion beams (350-KeV He^{+} simultaneously with 4.5-MeV Ni^{++}). TEM observations showed the formation of a high density of point-defect clusters and dislocation loops (<30 nm diameter) distributed uniformly in the specimens. Mechanical-property testing showed embrittlement of the alloy. TEM investigations of deformed microstructures were used to determine the causes of embrittlement and yielded observation of dislocation channels propagating through the undeformed matrix. Channels are the sole slip paths and cause early onset of necking and loss of work-hardening in this alloy. Based on a review of the available literature, suggestions are made for further research of slip localization in V-base alloys.

Extended Abstract

A systematic study has been initiated to evaluate the performance of several V-Cr-Ti alloys after exposure to environments containing hydrogen at various partial pressures. The goal is to correlate the chemistry of the exposure environment with the hydrogen uptake in the samples and its influence on the microstructure and tensile properties of the alloys. At present four heats of alloys (BL-63, BL-71, and

T87, plus 44 from General Atomics) are being evaluated. Other variables of interest are the effect of initial grain size on hydrogen uptake and tensile properties, and the synergistic effects of oxygen and hydrogen on the tensile behavior of the alloys. Experiments conducted thus far on specimens of various V-Cr-Ti alloys exposed to pH_2 levels of 0.01 and 3×10^{-6} torr showed negligible effect of H_2 on either maximum engineering stress or uniform/total elongation. Further, preliminary tests on specimens annealed at different temperatures showed that grain size variation by a factor of ≈2 had a negligible effect on tensile properties.

A systematic study was conducted to evaluate the oxidation kinetics of V-4Cr-4Ti ("44 alloy") and V-5Cr-5Ti alloys ("55 alloy") and to establish the role of oxygen ingress on the tensile behavior of the alloys at room temperature and at 500°C. The oxidation rate of the 44 alloy is slightly higher than that of the 55 alloy. The oxidation process followed parabolic kinetics. Maximum engineering stress for 55 alloy increased with an increase in oxidation time at 500°C. The maximum stress values for 55 alloy were higher at room temperature than at 500°C for the same oxidation treatment. Maximum engineering stresses for 44 alloy were substantially lower than those for 55 alloy in the same oxidation treatment. Uniform and total elongation values for 55 alloy were almost zero at room temperature after ≈500 h exposure in air at 500°C; the same values were 4.8 and 6.1%, respectively, at 500°C after ≈2060 h oxidation in air at 500°C. Uniform and total elongation values for 44 alloy were 1.6 and 3.6% at 500°C after 2060 h oxidation in air at 500°C.

Maximum engineering stress for 44 alloy at room temperature was 421.6-440.6 MPa after ≈250 h exposure at 500°C in environments with a pO_2 range of 1×10^{-6} to 760 torr. The corresponding uniform and total elongation values were 11-14.4% and 14.5-21.7%, respectively. Measurements of crack depths in various specimens showed that depth is independent of pO_2 in the preexposure environment and was of 70-95 µm after 250-275 h exposure at 500°C.

A systematic study has been initiated to examine the use of lasers to weld sheet materials of V-Cr-Ti alloys and to characterize the microstructural and mechanical properties of the laser-welded materials. In addition, several post-welding heat treatments are being applied to the welded samples to evaluate their benefits, if any, to the structure and properties of the weldments. Hardness measurements are made across the welded regions of different samples to evaluate differences in the characteristics of various weldments.

A series of tests is being conducted in the DIII-D tokamak to determine the effects of environmental exposure on a V-4Cr-4Ti vanadium alloy. These tests are part of the effort to build and install a water-cooled vanadium V alloy structure in the upgrade of the DIII-D radiative divertor. Data from the test series indicate that the performance

Microstructures of reduced-activation F81H (8Cr-2W-0.2V-0.04Ta) and the F82H steels doped with ^{10}B, irradiated at 250 and 300°C to 3 and 57 dpa in the High Flux Isotope Reactor (HFIR) have been examined by TEM. In the F82H irradiated at 250°C to 3 dpa, dislocation loops, small unidentified defect clusters with a high number density, and very few MC precipitates were observed in the matrix. The defect microstructure after 300°C irradiation to 57 dpa is dominated by the loops, and the number density of loops was lower than that of the F82H-^{10}B steel. Cavities were observed in the F82H-^{10}B steels, but the swelling value is insignificant. Small particles of M_6C precipitates were formed on the $M_{23}C_6$ carbides, which were present before the irradiation, in the F82H and the F82H-^{10}B by irradiation at 300°C to 57 dpa. A low number density MC precipitate was formed during irradiation at 300°C to 57 dpa in the matrix.

Charpy tests were conducted on eight normalized-and-tempered reduced-activation ferritic steels irradiated in two different normalized conditions. Irradiation was conducted in the Fast Flux Test Facility at 393°C to ≈ 14 dpa on steels with 2.25, 5, 9, and 12% Cr (0.1% C) with varying amounts of W, V, and Ta. The different normalization treatments involved changing the cooling rate after austenization. The faster cooling rate produced 100% bainite in the 2.25 Cr steels, compared to duplex structures of bainite and polygonal ferrite for the slower cooling rate. For both cooling rates, martensite formed in the 5 and 9% Cr steels, and martensite with ≈ 25% δ-ferrite formed in the 12% Cr steel. Irradiation caused an increase in the ductile-brittle transition temperature (DBTT) and a decrease in the upper-shelf energy. The difference in microstructure in the low-chromium steels due to the different heat treatments had little effect on properties. For the high-chromium martensitic steels, only the 5Cr steel was affected by heat treatment. When the results at 393°C were compared with previous results at 365°C, all but 5Cr and 9Cr steel showed the expected decrease in the shift in DBTT with increasing temperature.

Extended Abstract

The tensile properties of the three candidate alloys GlidCop™ Al25, CuCrZr, and CuNiBe are known to be sensitive to the testing conditions such as strain rate and test temperature. This study was conducted on GlidCop™ Al25 (2 conditions) and

TEM disk specimens of austenitic stainless steel 316LN irradiated to damage levels of about 3 dpa at irradiation temperatures of either about 90 or 250°C have been investigated by using transmission electron microscopy. The irradiation at 90 and 250°C induced a dislocation loop density of 3.5×10^{22} m^{-3} and 6.5×10^{22} m^{-3}, a black dot density of 2.2×10^{23} m^{-3} and 1.6×10^{23} m^{-3}, respectively, in the steels, and a high density ($<1 \times 10^{22}$ m^{-3}) of precipitates in matrix. Cavities could be observed in the specimens after the irradiation. It is suggested that the dislocation loops, the black dots, and the precipitates cause irradiation hardening, an increase in the yield strength, and a decrease in the uniform elongation in the 316LN steel irradiated at low temperature.

No contributions.

Postirradiation examinations of COBRA-1A beryllium pebbles irradiated in the EBR-II fast reactor at neutron fluences which generated 2700-3700 appm helium have been performed. Measurements included density change, optical microscopy, scanning electron microscopy, and transmission electron microscopy. The major change in microstructure is development of unusually shaped helium bubbles forming as highly non-equiaxed thin platelet-like cavities on the basal plane. Measurement of the swelling due to cavity formation was in good agreement with density change measurements.

Seven subcapsules from the FFTF/MOTA 2B irradiation experiment containing 97 or 100% dense sintered beryllium cylindrical specimens in depleted lithium have been opened and the specimens retrieved for postirradiation examination. Irradiation conditions included 370°C to 1.6×10^{22} n/cm^2, 425°C to 4.8×10^{22} n/cm^2, and 550°C to 5.0×10^{22} n/cm^2. TEM specimens contained in these capsules were also retrieved, but many were broken. Density measurements of the cylindrical specimens showed as much as 1.59% swelling following irradiation at 500°C in 100% dense beryllium. Beryllium at 97% density generally gave slightly lower swelling values.

Neutron fluence measurements and radiation damage calculations are reported for the joint U.S.-Japanese experiments JP-9, -12, and -15. These experiments were conducted in target positions of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) for a period of nearly four years. The maximum neutron fluence at midplane was 2.5×10^{23} n/cm^2 (7.1×10^{22} n/cm^2 above 0.1 MeV), resulting in about 60 dpa and 3900 apm helium in type 316 stainless steel.

Neutron fluence measurements and radiation damage calculations are reported for the joint U.S.-Japanese experiment JP-20, which was conducted in a target position of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). The maximum total neutron fluence at midplane was 4.2×10^{22} n/cm^2 (1.0×10^{22} n/cm^2 above 0.1 MeV), resulting in about 8.4 dpa and 388 appm helium in type 316 stainless steel.

Neutron dosimetry measurements have been conducted for various positions of the High Flux Beam Reactor (HFBR) at Brookhaven National Laboratory (BNL) in order to measure the neutron flux and energy spectra. Neutron dosimetry results and radiation damage calculations are presented for positions V10, V14, and V15.

The production of ^{4}He and tritium has been calculated for beryllium irradiated in the COBRA-1A2 experiment in the Experimental Breeder Reactor II. Reactor rates were based on adjusted neutron spectra determined from reactor dosimetry measurements at three different elevations in the region of the beryllium capsules. Equations are given so that gas production can be calculated for any specific capsule elevation.

The production of ^{4}He, ^{3}He, and tritium has been calculated for beryllium irradiated in the Materials Open Test Assembly (MOTA)-2B experiment in the Fast Flux Test Facility (FFTF). Reaction rates were based on adjusted neutron spectra determined from reactor dosimetry measurements at seven different elevations in the irradiation assembly. Equations are given so that gas production, dpa, and neutron fluences can be calculated for any specific elevation in the MOTA-2B assembly.

Extended Abstract

Neutron fluence measurements and radiation damage calculations are reported for the joint U.S.-Japanese experiment MFE-200-J-1, which was conducted in the removable beryllium (RB*) position of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). The maximum neutron fluence at midplane was 4.1×10^{22} n/cm^2 (1.9×10^{22} n/cm^2 above 0.1 MeV), resulting in about 12 dpa and 28 appm helium in type 316 stainless steel.

The DHCE (Dynamic Helium Charging Experiment) irradiation experiment was conceived to simulate fusion-relevant helium production in a fission reactor irradiation. The main objective is to maintain the Helium--to-DPA ratio at, roughly, the same level as expected in a fusion environment. The problem in fission reactor irradiation is that Helium production is very low, because the fission neutrons, for basically all structural materials relevant for fusion applications, do not have enough