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Fatigue, crack growth, and creep rupture properties were investigated in the context of lithium cooled first wall structural designs for commercial tokamak reactors. The studies resulted in the definition of the physical properties for fatigue, creep-fatigue, flaw growth, creep rupture, and irradiation swelling required to meet a goal life of 10 MW-Yr/m^2. Fatigue and flaw growth properties were shown to need improvements over presently defined properties for annealed Type 316 stainless steel.

The results of this study indicate that the use of refractory metals in the first wall, blanket, and header region of a fusion reactor offer economic advantage over stainless steel and titanium, provided they have modestly longer life or permit moderately a higher peak coolant temperature. If this use is expanded beyond the header, out through the primary coolant loop, the cost of electricity is significantly increased. This increase in cost is only recovered, relative to the use of stainless steel or titanium for a very narrow set of operating conditions. Therefore, it appears that the use of refractory metals should be restricted to the first wall, blanket, and header region only and stainless steel or titanium be used for the primary coolant loop.

Primary and backup materials were selected for an Experimental Power Reactor (EPR), Demonstration Power Reactor (DPR) and Commercial Power Reactor (CPR). Testing and manpower requirements for establishing the material property data needed for design was defined. Natural resources and industrial capability necessary to satisfy a fusion economy were examined. Structural material life, failure modes and radioactivity considerations were among the factors evaluated.

This study examines just how long one must contain radioactive titanium before it can be safety reprocessed. It was assumed that the spent first wall and blanket structural material would be completely reprocessed in a standard manufacturing facility capable of both primary and secondary fabrication.

It was found that reprocessing could occur when the chemical hazard associated with inhalation was greater than the hazard associated with inhalating the same amount of radioactive species. This conclusion allowed the use of the threshold limiting value (TLV) to set a limit on the airborne concentration of the elements. Then by calculating the time required for that amount of material to decay to the same diluent factor indicated by the biological hazard potential (BHP) in air, the time for reprocessing was determined. Based on these assumptions, it was determined that it is feasible to think of titanium, and some of its alloying elements as being recyclable in a relatively short time period.

Planning radiation damage experiments in fission reactors such as ORR, HFIR, and EBR-II in support of fusion reactor materials development requires for these facilities dpa and gas production rates for many potential materials. This report summarizes some of the calculations currently under way.

The geometry of a small fatigue specimen for irradiation testing has been defined. This specimen is fabricated from thin sheet material, occupies the same test volume as the SS-1 sheet tensile specimen, and increases the number of fatigue tests per unit volume of test space by a factor of 4 to 5.

Swelling in annealed type 316 stainless steel after HFIR irradiation at 580–780°C to fluences producing 1850–3300 at. ppm He and 30–47 dpa is reduced by the addition of 0.23 wt % Ti. This can he related to microstructural observation of finely distributed TiC precipitate particles, which accommodate helium in small cavities at the particle interfaces. Swelling in 20%-cold-worked type 316 stainless steel irradiated in HFIR at 580–590°C to fluences producing 30–80 at. ppm He and 1.5–3.0 dpa is reduced and recrystallization retarded by the addition of 0.23 wt % Ti.

Postirradiation tensile strength and ductility of annealed type 316 stainless steel after irradiation in HFIR at 580–780°C to neutron fluences producing 1850–4000 at. ppm He and 30–60 dpa are improved by the addition of 0.23 wt % Ti. These improvements correlate microstructurally with intragranular precipitates of TiC, which accommodate helium effectively, and smaller grain-boundary cavities above 600°C. Additional improvement is observed at 600°C when grain boundary $M_{23}C_6$ is produced during irradiation. Titanium addition to 20%-cold-worked type 316 stainless steel improved ductility and changes fracture mode *to* ductile transgranular after irradiation in HFIR at 580–600°C and to fluences producing 30–80 at. ppm He and 1.5–3.0 dpa.

Precipitation in annealed type 316 stainless steel after HFIR irradiation at 550–680°C to fluences producing 2000-3300 at. ppm He and 30–47 dpa is changed relative to fast reactor or thermal aging exposure to similar temperatures and times. The phases observed after HFIR irradiation are the same as those observed after aging to temperatures 70–200°C higher or for much longer times. There is a similar temperature shift in addition to different phases observed for HFIR irradiation compared with EBR-II. The changes observed are coincident with including simultaneous helium production to high levels in the irradiation damage products of the material.

Density measurements indicate significantly higher swelling in 20%-cold-worked type 316 stainless steel irradiated in HFIR than in EBR-TI. Of the three temperatures investigated – 375, *475,* and 575°C highest swelling occurred at 375°C.

Weld-containing samples irradiated in HFIR at about 55°C exhbited large increases in the 35°C tensile strength but only modest loss of ductility. The strength of the weld-containing samples remained below the strength of the base metal, and all tested samples failei within the weld metal.

Irradiation of 20%-cold-worked type 316 stainless steel at -55°C to produce 4.5 to 10.5 dpa and 180 to 520 at. ppm He reduces the uniform elongation to near zero but has little effect on the total elongation or strength properties for tensile tests at 35°C. Tests in the temperature range 200 to 300°C show the same trends as for 35°C tests. Tests at 400 to 600°C show elongation and strength very close to the control values. Tensile testing at 700°C after holding the sample 30 min at temperature shows low ductility but strength values equal to the control values. This appears to be the onset of helium emhrittlement.

Initial experiments have begun on the effects of HFIK irradiation of PE-16, where large amounts of helium are produced in the alloy. Irradiation at 400 and 650°C produced cavities coated with the γ' precipitate phase. After irradiation at 400°C and a helium content of 370 at. ppm, the average cavity size was 16.5 nm; at 650°C and a helium content of 1030 at. ppm, a bimodal cavity distribution was created with average sizes of 31.3 and 124.0 nm. Faulted loops were formed at 400°C but not at 650°C. A new observation for irradiated PE-16 was the precipitation of $M_{23}C_6$ in the grain boundaries at 650°C. Experiments under way should help explain many of the microstructural changes in these exploratory PE-16 specimens.

Four alloys were selected for the scoping studies and they represent the three major classes or divisions, of titanium alloys, alpha, beta, and alpha plus beta. In addition, three heat treatments were selected to determine the effect of microstructure on radiation damage. The identified alloys and their heat treatments have been reviewed by the titanium community and their comments noted, A titanium inventory has been established consisting of the selected alloys and their heat treatments.

A large data base has been compiled on vanadium-base alloys but the data base on any one alloy is quite limited. Great flexibility exists in the composition-microstructure-property relationship and this facilitates alloy optimization to meet diverse property requirements. Tensile properties and creep properties of existing alloys exceed likely requirements. Fatigue strength, including crack growth rate, is probably the most critical material property but no data exists for vanadium alloys. Swelling and irradiated ductility behavior look promising but require further evaluation. Vanadium alloy-liquid metal compatibility, particularly interstitial mass transfer, may be equally as critical as fatigue behavior; viability cannot be established with the existing data base. Fabricability must be given early consideration in alloy selection to guard against potentially serious problems in subsequent scale-up and production.

A Ti-6AL/4V torsion creep specimen was irradiated with 16 MeV protons Lo a total dose of 0.35 dpa. The Lest was conducted in a helium atmosphere at a nominal irradiation temperature of 325°C. The wire specimen was stressed Lo 138 MPa (maximum shear) throughout a preirradiation thermal creep period and during the first 50 hours of irradiation. Following the first 50 hours of irradiation, the stress was reduced to zero for the subsequent 4 hours of irradiation.

The results of this single test indicate that the irradiation creep rate in Ti-6AL/4V is within the same order of magnitude as that of similarly irradiated 20% CW 316 SS specimens. Results of the stress reduction show that the recovered strain is similar in magnitude to that which occurred during the initial transient and is irradiation dependent. A number of mechanisms describing the recovery are discussed.

Irradiation of niobium and Nb-1% Zr followed by tensile tests at about 35°C has shown no intrinsic difference in the tensile behavior of these two materials, The onset of plastic instability in both occurs at about 0.1 dpn, with strengthening continuing for irradiation beyond at least 0.36 dpa. Although the uniform elongation goes to near zero, the total elongation remains high, and the fracture mode is fully ductile. Ductility loss through plastic instability is related to deformation by dislocation channeling.

The hydrogen permeability of Inconel 625 was measured over a range of temperatures and hydrogen pressures that included values relevant to the Doublet III experiment (Inconel 625 is being used *as* the vacuum vessel material for Doublet III). The permeation rate obeyed a half-power dependence on hydrogen pressure down to the lowest pressure studies (10^{-1} Pa). The activation energy and permeation coefficient were in good agreement with published data for other Inconel-type alloys. These results will also be useful as a standard for comparison of the effectiveness of permeation barrier methods applied to path B alloys.