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# Alloy Development for Irradiation Performance

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*316 @ 55°C - 89*

*PE -16 - 96*

*WIFFEN 110 - 142*

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        Commercialization of Tokamak-Based Fusion Reactors (ANL and  
        McDonnell Douglas) . . . . . 2

        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<sup>2</sup>. Fatigue and flaw growth properties were shown to need  
        improvements over presently defined properties for annealed Type 316  
        stainless steel.

    1.2 Economic Impact of Using Refractory Metals for Fusion Reactors  
        (McDonnell Douglas) . . . . . 8

        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 tempera-  
        ture. 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. There-  
        fore, 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.

    1.3 Alloys for the Fusion Reactor Environment – A Technical Assessment  
        (McDonnell Douglas and ADIP Task Group) . . . . . 14

        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.

1.4 Recycling Potential of Titanium Alloys (McDonnell Douglas, University of Wisconsin, and BNL) . . . . . 23

This study examines just how long one must contain radioactive titanium before it can be safely 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 inhaling 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.

1.5 Calculation of Irradiation Response Rates for Fission Reactors Used by the ADIP Irradiation Programs (ORNL) . . . . . 30

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.

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3. PATH A ALLOY DEVELOPMENT - AUSTENITIC STAINLESS STEELS . . . . . 39

3.1 Comparison of Titanium-Modified and Standard Type 316 Stainless Steel Irradiated in HFIR - Swelling and Microstructure (ORNL) . . . . . 40

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 be 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.

3.2 Mechanical Properties of Type 316 and Titanium-Modified Type 316 Stainless Steel Irradiated in HFIR (ORNL) . . . . . 54

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.

3.3 Precipitation Response of Annealed Type 316 Stainless Steel in HFIR Irradiations at 550 to 680°C (ORNL) . . . . . 63

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.

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ductility. The strength of the weld-containing samples remained below the strength of the base metal, and all tested samples failed within the weld metal.

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Initial experiments have begun on the effects of HFIR 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  $\gamma'$  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.

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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.

5.2 Technical Assessment of Vanadium-Base Alloys for Fusion Reactor Applications (Westinghouse) . . . . . 119

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.

5.3 16 MeV Proton Creep in a Ti-6Al/4V Specimen (HEDL) . . . . . 128

A Ti-6Al/4V torsion creep specimen was irradiated with 16 MeV protons to a total dose of 0.35 dpa. The test was conducted in a helium atmosphere at a nominal irradiation temperature of 325°C. The wire specimen was stressed to 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.

5.4 Plastic Instability in Neutron-Irradiated Niobium Alloys (ORNL) . . . 142

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 dpa, 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.

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