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# Alloy Development for Irradiation Performance

Quarterly Progress Report  
For Period Ending June 30, 1979

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**U.S.** Department of Energy  
Assistant Secretary for Energy Technology  
Office of Fusion Energy

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<p style="margin-left: 40px;"><i>The Materials and Radiation Effects branch of the Office of Fusion Energy is in the process of developing a Materials Handbook for Fusion Reactor Systems (MHFS). The handbook will be similar in format to the Nuclear Systems Materials Handbook developed for the breeder reactor program but will contain materials and material properties that are relevant to the fusion program. The contents of the MHFS will initially be limited to the first wall and blanket structural materials, and will be made available to those involved in fusion design and system studies. Based on information received from key personnel currently working on the ETF, INTOR, and the commercial tokamak projects, the structural materials of most interest to them are 20% cold worked 316 stainless steel, ferritic alloys, and titanium alloys. Therefore, for the near term (next 6 months) efforts will be directed to develop data sheets on these alloys for inclusion in the handbook.</i></p>		
1.2 Ferritic Stainless Steels for Fusion Applications (General Atomic Company) . . . . .		a
<p style="margin-left: 40px;"><i>The applications base and performance history of 9-12 Cr steels pertinent to use in fusion reactors was assessed. The most significant long-term high temperature usage has been in European fossil-fired power plants, where performance of welded structures has been excellent. A detailed metallurgical evaluation by General Atomic Co./Sulzer Brothers of an HT-9 superheater tube after 80,000 hours of service indicated good thermal stability, excellent residual mechanical properties, and reduced but significant impact toughness.</i></p> <p style="margin-left: 40px;"><i>The key issues for the application of these steels in fusion systems were identified as the adequacy of fracture resistance, performance in the fusion irradiation environment, and the acceptability of fabrication/welding characteristics. The results of a data base assessment of candidate alloys, commercial UT-9 (12 Cr-1 Mo) and developmental 9 Cr-1 Mo, indicated significant need for irradiation performance data.</i></p>		
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*Elevated temperature fatigue crack growth data has been generated using an electrical potential technique of monitoring crack extension in miniature center-cracked-tension specimens. The technique produced reliable accurate results at 316 and 260°C when compared with data generated from conventional compact tension specimens.*

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*The design of an in-reactor fatigue machine capable of performing tension/tension cycling on a chain of center-cracked-tension specimens in the ORR is nearing completion. A tentative test matrix has been established using the Path A Reference Alloy, 20% cold work 316 stainless steel.*

2.3 Neutronic Design of Spectral Tailoring Experiments (ORNL) . . . , . . . . . 35

*We have calculated the effect of a tantalum core piece in the ORR on the thermal neutron flux and displacement rate in an enclosed experimental capsule. The large thermal capture cross section of tantalum [about  $2.0 \times 10^{-27} \text{ m}^2$  (20 b)] reduces not only the thermal flux but also the displacement rate in the experimental capsule as a result of localized reduction in the number of fissions. This reduction in displacement rate can be partly offset by increasing the fuel loading in adjacent core pieces.*

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*We have designed and drawn the two experiments ORR-MFE-4A and ORR-MFE-4B. The first experiment will contain specimens at 300 and 400°C, and the second will contain specimens at 500 and 600°C. Parts are being fabricated for a bench test of the experiment, and a gamma heat measuring experiment is awaiting irradiation.*

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*Specimens of 20%-cold-worked type 316 stainless steel were irradiated in the High Flux Isotope Reactor (HFIR) to fluences of 0.85 to  $1.9 \times 10^{26} \text{ n/m}^2$ , resulting in displacement levels of 7 to 15 dpa and helium contents of 250 to 860 at. ppm. Fully reversed strain controlled fatigue tests were performed with strain ranges of 0.40 to 2.0%. The specimens were irradiated and tested at 430°C and tested in a vacuum at a pressure below  $10^{-4} \text{ Pa}$ . The fatigue life was reduced by factors of 3 to 10. We are studying fracture surfaces and will subsequently report the results.*

3.2 Microstructural Design for Fusion First-Wall Applications and Recommendations for Thermal-Mechanical Preirradiation Treatments (ORNL) . . . . . 48

*Previous work [including High Flux Isotope Reactor (HFIR) tests] on the effects of helium in metals indicates that the chances of maintaining good mechanical response in a fusion reactor depend on keeping helium away from the grain boundaries and trapping it intragranularly. Swelling from helium bubble formation is inevitable, but the finest helium distribution will minimize swelling. We propose a matrix of desirable preirradiation microstructures for the PCA based upon responses of type 316 stainless steel or type 316 modified with titanium to HFIR irradiation. Desirable microstructures result from a combination of grain boundary precipitation, cold work, and distribution of titanium-rich MC. We also discuss thermal-mechanical treatments to produce these microstructures. Further work includes completing the fabrication procedures on which microstructural homogeneity and control intimately depend.*

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*We studied the morphology and composition of mass transfer deposits in lithium-type 316 stainless steel thermal-convection loops (TCLs). Initially the deposits were needle-like crystals of nearly pure chromium. After longer operating times the deposits were blocky, and their composition varied as a function of their distance above the original surface. The later-formed deposits contained iron, chromium, and nickel. However, they were enriched in chromium and nickel relative to the concentrations of these two elements in the stainless steel. A plug that formed in a TCL after 9000 h consisted of lithium and a tangle of chromium crystals.*

8.3 Hydrogen Dissolution and Permeation Characteristics of Titanium-Base Alloys (Argonne National Laboratory) . . . . . 95

*The hydrogen permeation characteristics of an anodized sample of Ti-6Al-4V were measured from 450 to 665°C for hydrogen driving pressures in the range from 0.2 to 1.7 Pa. The permeability of the anodized sample was found to be about the same as that of the ion-nitride coated sample of Ti-6Al-4V studied previously, and about an order of magnitude lower than that of pure Ti-6Al-4V. A least-squares analysis was conducted on data for pure Ti-6Al-4V, ion-nitride-coated Ti-6Al-4V, and anodized Ti-6Al-4V wherein the preexponential term and the activation energy were varied independently for each sample but a common refined value for the pressure exponent was sought. It is clear from the results of this analysis that the coating procedures caused a measurable but far less than desirable reduction in the permeability of Ti-6Al-4V. The optimized pressure exponent, 0.64, suggests that the normally-assumed bulk-diffusion-limited permeation mechanism is being affected or perhaps totally superseded by another mechanism involving the surfaces of the alloy.*

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The stainless-steel-clad *vanadium-15% chromium* alloy loop for studies of circulating liquid lithium has continued to operate through the third quarter of FY-1979. This 0.5-liter-capacity, forced circulation loop is being used in investigations of (1) the distribution of non-metallic elements in *lithium/refractory metal* systems and (2) effects of a lithium environment on the mechanical properties of selected refractory metals. Exposure of *zirconium* foil to lithium at both 673 and 873 K resulted in significant pickup of nitrogen by the zirconium and a corresponding reduction of the nitrogen concentration in *lithium*.