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Many studies have suggested that Fe-Ni and Fe-Ni-Cr alloys of ~36 wt % Ni are susceptible to decomposition into two FCC phases. Ye herein model the kinetics and thermodynamics of spinodal deconposition, nucleation, growth, and coarsening of these alloys. Prediction of the scale of a microstructure formed by growth or coarsening may be done with some confidence. The absence of a model for surface anergy and gradient energy in systems with negative departure from ideality reduces us to little more than guesses far the kinetics of both nucleation and of spinodal deconposition. Ye can combine our calculations with those of others and studies of terrestrial and celestial decomposition to reach a conclusion that suggests a miscibility gap for Fe-Ni alloys in the Invar regime. The gap is found to be very narrow at high temperatures but to be broadened at low temperatures by magnetic effects.

Microstructures of Fe-15Cr-XNi (X = 15, 19, 21, 30, 35, 45, 75) ternary allays and a 15Cr-85Ni binary alloy irradiated in the EBR-II reactor at 510 and 538°C to various neutron fluences *were* examined by TEN. At 510°C void nucleation was complete prior to the onset of steady-state swelling at all nickel levels, implying that some process other than vold nucleation determines when the relatively rapid void growth associated with the onset of steady-state swellIng occurs. The protracted loop dominance of dislocation structure and the relatively low void density at intermediate nickel levels (30-45 wt %) suggested a reduced dislocation bias toward preferential interstitial absorption as a possible mechanism to produce lower swelling in this nickel range.

The present effort is intended to describe microstructural response to fast neutrons of two tungsten stabilized ferritic/martensitic steels and HT-9 following irradiation to doses as high as 100 dpa. The major change in microstructural development at 420°C from 34 to 103 dpa was found to be due to void evolution. No significant differences in dislocation or precipitate Structures could be identified, whereas void structures were found to have developed more uniformly and the resultant swelling was greater. The Precipitate structure that had formed during irradiation to 34 dpa remained stable to higher dose. Precipitate distributions showed only modest increases in size, indicating that this phase has stabilized. and will not change significantly with further irradiation. Therefore. tungsten stabilized martensitic steels appear to have excellent irradiation resistance, and can be expected to perform well to doses significantly higher than 100 dpa.

Steady-state swelling of HTO and 9Cr-1Mo in FFTF/MOTA appears to develop after 75 dpa for irradiation temperatures in the range 403-410°C. The rats of Swelling in the stress-free condition is very low, on the order of 0.015%/dpa. steady-state creep rates also develop concurrently, with creep not only proportional to the applied stress above a certain level but also to the swelling rate. The creep-swelling coupling coefficient for these ferritic/martensitic alloys appears to be very close to that of austenitic alloys. At 520°C the creep rate is linear with stress at all stress levels but at 600°C these alloys exhibit stress exponents greater than unity, reflecting the onset of thermal creep possibly activated by phase instabilities.

The microstructure of 9Cr-1MoVNb steel wolves quite differently during thermal aging and during neutron irradiation. During thermal aging at temperatures of 482 to 600°C for times up to 25.000 h. the as-tempered microstructure of lathlsubgrain boundaries and carbides ($M_{23}C_6$ and MC) remains unchanged while films and particles of Laves phase form along grain and subgrain boundaries. and dense dispersions of fine VC needles form in the matrix. By contrast, High Flux Isotope Reactor (HFIR) irradiation at 300 to 500°C to 37 to 39 dpa (about 10,000 h) causes the as-tempered subgrain boundary and carbide precipitate structure to become unstable, but does not produce precipitation of any new phases. Irradiation in HFIR causes partial dissolution of $M_{23}C_6$, coarsening and compositional evolution of MC, and recovery and coarsening of the lathlsubgrain structure.

Irradiation of ferritic steels with neutrons in the temperature range of room temperature to ~450°C results in lattice hardening, which causes an increase in strength and a decrease in ductility. Eight reduced-activation Cr-W-V steels with Chromium concentrations varying from 2.25 to 12% were irradiated at 365°C to ~7 dpa in the Fast Flux Test Facility. Steels containing the combination of 2% W and 0.25% V hardened less than those containing vanadium or tungsten alone or a steel with 1% W and 0.25% V. The amount of hardening was similar for Cr-W-V steels with constant vanadium and tungsten concentrations and containing 2.25, 5 and 9% Cr; a steel with 12% Cr hardened considerably more. Specimens of two conventional Cr-Mo steels. 9Cr-1MoVNb and 12Cr-1MoVW, were also Irradiated. and the hardening of these steels was similar to analogous reduced-activation steels.

The effect of heat treatment on the impact behavior of eight experimental heats of reduced-activation ferritic steels was investigated. Steels with 2 1/4, 5, 9, and 12 wt % Cr and containing tungsten, vanadium, and tantalum were examined. Impact properties of steels with 2 1/4% Cr depended on microstructure. which was affected by cooling rate after austenitization. By heat treating the 2 1/4Cr steels to change the microstructure from a bainitic structure containing ferrite to one without ferrite, the ductile-brittle transition temperatures were reduced substantially. Cooling rate had essentially no effect on the high-Chromium martensitic steels.

The irradiation-induced creep and swelling of AISI 304L in EBR-II at ~390°C have been investigated to exposures on the order of 80 dpa and compared with the behavior of AISI 316 stainless steel. It is shown that swelling and creep of various austenitic steels are strongly interactive phenomena. While Swelling depends on stress, displacement rate, composition and cold-work level, the creep rate directly depends only on the stress level and the instantaneous swelling rate. The creep-swelling coupling coefficient does not appear to be very sensitive to composition, cold-work level or temperature.

The candidate austenitic alloy designated PCA was irradiated in FFTF/MOTA in the 20% cold-worked condition to fluences ranging as high as 122 dpa. At temperatures in the range of 384–550°C the creep rate was found to be linearly dependent on the applied stress level. The assumption that the $B_0 + DS$ creep model is applicable to this alloy yields a creep-swelling coefficient of $D \approx 0.6 \times 10^{-2}\ MPa^{-1}$, in excellent agreement with the results of studies on solution annealed AISI 304L and various thermomechanical treatments of AISI 316 stainless steel.

The development of alloys With reduced long-life radioactivity has been incorporated into the goals of fusion materials development. One possible candidate alloy is an Fe-12Cr-6.5Mn-1W-0.3V-0.1C stabilized martensitic steel. The phase stability of this steel has been investigated following neutron irradiation over a temperature range of 420 to 600°C to doses as high as 100 dpa. The microstructural response of the irradiated steel has been compared to that of the same alloy aged for 10,000 h from 365 to 600°C. $M_{23}C_6$ was the only carbide found in both the aged and irradiated Structures. In the samples aged at 420 and 520°C, and the samples Irradiated at 520°C, an Fe-Cr-Mn-W chi phase formed. In the samples Irradiated at 420°C, alpha prime formed. The presence of chi phase in both the thermally aged and neutron Irradiated steels indicates that the Fe-Cr-Mn system is prone to intermetallic phase formation so that careful study on the effects of alloying are needed.

An Fa-Cr-Mn-W chi phase formed in a 12Cr-6.5Mn-1W-0.3V-0.1C (wt %) martensitic steel that was aged from 1000 to 5000 h at 420 and 520°C. An Fe-25%Cr-15%Mn alloy with a conposition close to that of the chi phase was fabricated and aged for 2000 and 5000 h in order to determine phase stability. Optical and transmission electron microscopy of the samples showed that sigma, ferrite, austenite and $M_{23}C_6$ formed. These results show that chi phases does not form easily in simple Fe-Cr-Mn alloys and that chi may require minor additions of tungsten or molybdenum to form thermally.

The phase stabllity of manganese-stabilized austenitic alloys based on Fe-12Cr-20Mn-0.25C with minor additions of W, Ti, V, P, and B has been evaluated at 600°C. Dominant precipitate phases in both annealed and cold-worked materials are $M_{23}C_6$ and TiC carbides, With no intermetallic phases being observed. Conposition differences between matrix and grain boundary $M_{23}C_6$ were observed in aged specimens. Those differences are associated with the formation of grain boundary $M_{23}C_6$ during annealing at high temperature. Minor alloy additions have Significant influence on the precipitate structure and Conposition. Tungsten is strongly incorporated into $M_{23}C_6$ and TiC, Whereas vanadium is not. Phosphorus is incorporated into $M_{23}C_6$ but not into TiC. The austenite rtablllity even after the precipitation of carbides during high temperature annealing indicates that the carbon level of there alloys can be reduced and still maintain a stable austenite matrix with good resistance to intermetallic formation.

Helium was inplanted in type 316 stainless steel, through tritium decay, to levels of 0.18, 2.5, 27, 105, and 256 appm. Bead-on-plate welds were then made using the gar tungsten arc process. Intergranular cracking occurred in the heat-affected zones (HAZ) of specimens with helium concentrations equal to or greater than 2.5 appm. No such cracking was observed in helium-free control specimens or in specimens containing the lowest helium concentration. In addition to the HAZ cracking, brittle. centerline cracking occurred in the fusion zone of specimens containing 105 and 256 appm He. Transmission and scanning electron microscopy results indicated that both the HAZ cracking and centerline cracking in the fusion zone resulted from the stress-induced growth and coalescence of Cavities initiated at helium bubbles on interfaces. For the HAZ case. the cavity growth rate is modeled and is shown to predict the experimentally measured 1 s time lag between peak weld temperature and the onset of cracking.

A number of simple vanadium alloys have been examined by transmission electron microscopy following Irradiation in the Materials Open Test Assembly of the Fast Flux Test Facility (FFTFIMOTA) at 600°C to 14 dpa. The alloys included pure vanadium. V-4%Mo, V-9%W, V-10 and V-15%Cr, V-5, 10, and 20%Ti, V-20%Ti-5%Y (in welght percent) and VANSTAR-7. All specimens were found to have developed features typical of radiation damage. However, large differencar were found as a function of composition so that void, dislocation and precipitate structures were dissimilar even for specimens of similar composition. For example, different heats of pur. vanadium with different interstitial solute contents were examined. One contained voids and the other did not. In a series of three V-Ti alloys. precipitation varied as a function of titanium content. However, in general, specimens displayed precipitate development. Often indicated by {001} stacking fault features. About half the alloys contained voids. Dislocation structures included large loops and dislocation tangles but much smaller loops similar to black spot damage could be frequently identified. Therefore, microstructures developed under irradiation are very sensitive to initial conposition.

Dehydrogenation of V-15Cr-5Ti Charpy inpact specimens after irradiation at 400°C to 114 dpa was performed prior to inpact testing. Hydrogen removal caused no change in the inpact behavior of the alloy, yielding a DBTT In excess of 400°C.

Recent results have indicated that modest levels of neutron irradiation damage in V-15Cr-5Ti lead to severe degradation of mechanical properties. In order to explain the observed behavior. specimens of V-15Cr-5Ti have been examined by transmission electron microscopy following irradiation at 600°C to 14 dpa in the Fast Flux Test Facility (FFTF). The conditions examined included two heats of V-15Cr-5Ti with very different oxygen contents and, in one case, a section of a Charpy specimen demonstrated to have very poor inpact resistance. Specimens irradiated at 420 and 520°C were also examined to confirm density change measurements that indicated swelling levels as hlgh as 2.5%. Following irradiation at 600°C. a complex microstructure was found that included a perfect dislocation network of line segments and loops. faulted loops, a high density of black spot damage and a high density of rod-shaped precipitates. Heat-to-heat variations did not appear to significantly alter these microstructural features: instead the density of large blocky precipitates, believed to be TiO_2, was found to have increased with increasing oxygen content. Only two of the specimens irradiated at 600°C contained voids, whereas none of the specimens irradiated at 420 or 520°C contained significant void swelling. Therefore, swelling as measured by density change is probably due to phase instability in there specimen conditions, and the hardening and irradlatlon embrittlement that have been observed can be explained by the microstructural development. Phase stability appears to be an important criterion in the design of vanadium alloys for fusion applications.

The tensile properties of V-13.5Cr-5.2Ti, V-9.2Cr-4.9Ti, V-7.2Cr-14.5Ti, V-3.1Ti-0.3Si, and V-17.7Ti alloys were determined after neutron irradlatlon at 420, 520, and 600°C to irradiation damage levels ranging from 21 to 07 dpa. These alloys exhibited a maximum of irradlatlon hardening in the range of 25 to 50 dpa. These alloys also exhibited significant ductility (>3%) after irradiation at 420, 520, and 600°C to 07 dpa.

The tensile and fracture behavior of a variety of dispersion strengthened copper alloys were examined after irradiation at 411, 414, and 429°C to doses ranging from 32 to 50 dpa. The internally oxidized Glidcop alloys appear to be the best candidates for high heat flux service. Unfortunately. laser welding completely destroys their favorable properties.

The changes observed in electrical conductivity of pure copper in a variety of neutron irradlatlon experiments at ~400°C are quite consistent. The addition of relatively minor levels of solutes doer not appear to change the Irradiation response of the alloys beyond the changes observed in the unirradiated condition.

The intrinsic swelling rate of pure copper and Cu-5Ni appears to be ~0.5%/dpa at ~400°C. At 529°C the swelling rate of these alloys appears to be much loner. In more complex precipitation-strengthened alloys. phase instabilities appear to play a role in determining the onset of void swelling and also the density of the unvoided matrix. Dispersion-hardened alloys were found to be very resistant to swelling. Large amounts of oxygen in an alloy, however, can overcome the swelling resistance imparted by very hlgh densities of oxide dispersoids. Internally oxidized Glidcop alloys offer the most pronounced resistance to swelling. with the translent regime progressively extended as alumina content increases. Unfortunately. laser welding destroys the swelling resistance of dispersion-strengthened alloys.

The electrochemical potentiokinetic reactivation (EPR) test technique was applied to the determination of sensitization in a neutron-irradiated (420°C, 10 dpa) PCA. Miniaturized specimens (3 mm diam by 0.25 mm thick) in both solution-annealed and 25% Cold-worked conditions were tested and the degree of sensitization (DOS) was Calculated in terms of the reactivation Charge. Pa. Results indicated the presence of radiation-induced sensitization as compared to control specimens thermally aged at the irradiation temperatures. Post-EPR test examination of the specimen surfaces showed etching across the face of each grain as well as grain boundaries. This indicates that the Pa value normalized by total grain boundary area, which is an accepted EPR-DOS criterion to determine the susceptibility of thermally sensitized stainless Steels to intergranular stress corrosion cracking (IGSCC). is not directly applicable to radiation-induced sensitization for the condition investigated. Further Investigations are necessary to Correlate the results in this study to the IGSCC susceptibility of the irradiated stainless steel.

Chemical interactions involving carbon have a dominant role in the corrosion behavior of ferrous and vanadium-base alloys in lithium. Experimental investigations with these alloys in austenitic forced-circulation lithium loops have shown that weight changes in these systems are influenced by both mass transfer and deposition. The carburization-decarburization behavior of both commercial and high-purity experimental austenitic and ferritic alloys in flowing lithium has been conpared with that observed in flowing sodium at temperatures of 550 to 700°C. Type 316 stainless steel (SS) and various experimental and commercial heats of Fe-Cr-Mo ferritic materials were investigated to determine the effect of varying levels of chromium and molybdenum. Carbon transfer was evaluated in both pure vanadium and vanadium-base alloys containing various amounts of chromium and titanium, such as V-10Cr-3Fe-Zr, V-15Cr-5Ti, V-10Ti, and V-20Ti.

The SCC susceptibility of Types 316NG, 316. and 304 stainless steel (SS) has been investigated in slow-strain-rate tests (SSRTs) in aqueous environments that simulate important parameters anticipated for ITER first wall/blanket systems. Initial SSRTs were performed on both crevice and noncrevice specimens in oxygenated water with sulfate additions at a strain rate of $3 \times 10^{-7}\ s^{-1}$ and temperatures of 95 and 150°C. Scanning electron microscopy showed no evidence of SCC. This was substantiated by an evaluation of the load-carrying capability of identical specimens of Type 316NG SS in air and Water. The stress ratio, which is an indicator of cracking susceptibility. was defined as the ratio of the increase in stress after local yielding in the environment to the Corresponding stress difference in an identical test in air, both computed at the same strain. A ratio of ~1.0 is indicative of a low SCC susceptibility, which was the situation in our initial experiment on Type 316NG SS at 95°C.

Standard chemical immersion (modified Strauss) tests and analytical electron microscopy (AEM) showed that reduced activation austenitic Fe-Hn-Cr Steels based on Fe-20Mn-12Cr-0.25C (wt %) are extremely prone to thermal sensitization and resulting intergranular corrosion because of their high carbon contents and low chromium concentrations. Therefore. this susceptibility to sensitization after appropriate thermal aging, fabrication, or irradiation makes their use in aqueous and certain other environments problematical. Excellent correlation between intergranular corrosion induced by immersion in the acidified $CuSO_4$ solution and the presence of narrow chromium-depleted Zones around grain boundaries, as determined by AEM. was found. Due to the need to meet reduced activation requirements. the opportunities to increase the Sensitization resistance of fully austenitic Fe-Mn-Cr steels by alloy design are limited.

Sheet tensile specimens of commercial cast beryllium are found to be brittle following irradiation at 400°C to 8 dpa. Failure is by either transgranular brittle fracture or grain boundary embrittlement, depending on specimen thickness. Embrittlement is ascribed both to helium bubble formation at grain boundaries and to C-type loop formation. Loop formation of this type has not previously been reported in the literature.

Tritium release kinetics have been measured for neutron-irradiated beryllium in the temperature range of 573-884 K. Two tritium release tests of 270 and 550 h are the first in a series of tritium release tests on irradiated Be in support of the International Thermonuclear Experimental Reactor (ITER) blanket designs. Real-time tritium release curves were measured by an ionization chamber after temperature step changes of 100 K under tightly controlled conditions specific to ITER. The experimental conditions were designed to measure diffusion-controlled release of tritium from irradiated Be. The preliminary conclusion from this data set is that a unique rate-controlled mechanism is not distinguishable. Apparent diffusion kinetics at lower temperatures give way to a burst release at 884 K, coincident with a restructuring and bubble coalescence in this fully dense material.