Nuclear Engineering Division

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Corrosion and Mechanics of Materials

Light Water Reactors

Irradiation-Induced Stress Corrosion Cracking of Austenitic Stainless Steels

In recent years, failures of reactor internal components have been observed after the components have reached neutron fluence levels > 5 x 10^{20} $n \cdot cm^{-2}$ (E > 1 MeV). The general pattern of the observed failures indicates that as nuclear plants age and fluence increases, various apparently nonsensitized austenitic stainless steels become susceptible to intergranular failure. Some components (e.g., BWR core shroud, control-blade handle and sheath) have cracked under low applied stresses. Although some failed components can be replaced, structural components such as the BWR top guide, shroud, or core plate would be very difficult or impractical to replace. Understanding IASCC is thus necessary to determine operating and inspection requirements for reactors that have reached threshold fluence levels.

The primary effects of irradiation on the reactor internals, which are usually fabricated from Type 304, 316, or 348 SSs, include alteration of microchemistry, microstructure, and mechanical properties. Irradiation produces defects and defect clusters in grain matrices and alters the dislocation network, dislocation loop, and dislocation channel structures, leading to radiation-induced hardening. Irradiation also leads to changes in the stability of second-phase precipitates and the local chemistry near grain boundaries, precipitates, and defect clusters. Grain-boundary microchemistries that differ significantly from the bulk composition can be produced not only by radiation-induced segregation but also by thermally driven equilibrium and nonequilibrium segregation of alloying and impurity elements. Neutron irradiation also alters the water chemistry. Although irradiation-induced grain-boundary depletion of Cr has been considered for many years to be the primary metallurgical process that causes IASCC, numerous IASCC characteristics cannot be explained well by Cr-depletion theory.

Our work in this area includes:

a. determination of crack growth rate, fracture toughness, and susceptibility to IASCC of irradiated stainless steels in BWR-like water;
b. stress corrosion testing, tensile testing, and microstructural analysis of decommissioned EBR-II (Experimental Breeder Reactor-II) components irradiated to very high dose;
c. microstructural characterization and failure analyses of BWR core shroud welds and simulated mockup welds fabricated by a shielded-metal-arc procedure; and
d. test design and specimen irradiation in a PWR-relevant irradiation experiment in the BOR-60 reactor in Russia, in cooperation with the International Group on Cooperative IASCC Research.

Highlights of these efforts are summarized below.

Crack Growth Rates and Fracture Toughness of Austenitic Stainless Steels in BWRs

Austenitic SSs are used extensively as structural alloys in the internal components of reactor pressure vessels because of their superior fracture toughness. However, exposure to high levels of neutron irradiation for extended periods leads to significant reduction in the fracture resistance of these steels. Experimental data have been obtained on fracture toughness, corrosion fatigue, and SCC of austenitic SSs that were irradiated to fluence levels of ≈0.3, 0.9, and 2.0 x 10^{21} n/cm^2 (E > 1 MeV) (≈0.45, 1.35, and 3.0 dpa) at ≈288°C. The irradiations were carried out in a helium

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environment in the Halden heavy water boiling reactor. Fracture toughness tests were conducted in air, and CGR tests in normal water chemistry (NWC) and low-DO BWR environments at 288°C.

Neutron irradiation at 288°C decreased the fracture toughness of all steels tested. ***Figure a*** gives the fracture toughness curves for Type 304 SS derived from tests at Argonne, the Japan Power Engineering and Inspection Corp. (JAPEIC), and General Electric (GE). The data from commercial heats (JAPEIC and GE) generally fall within the scatter band (dashed lines in figure) for the data obtained at higher temperatures. For Type 304 SS irradiated to 0.3, 0.9, and 2.0 x 10^{21} n/cm^2, the fracture toughness values decrease as follows: 507, 313, and 188 kJ/m^2, respectively. The fracture toughness curves for irradiated Types 304 and 316 SS are comparable.

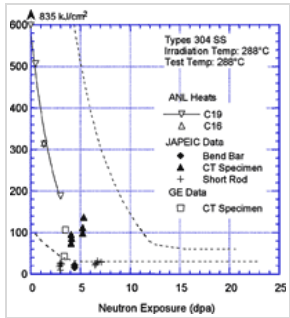


Figure a: Fracture toughness (JIc) of austenitic stainless steels as a function of neutron exposure at 288°C. Click on image to view larger image.

The tests for the steels irradiated to either 0.9 or 2.0 x 10^{21} n/cm^2 indicate significant enhancement of CGRs in the NWC BWR environment (see ***Figure b***). The CGRs under SCC conditions are a factor of ≈5 higher than the disposition curve proposed in the NRC report NUREG–0313 for sensitized austenitic SSs. In low-DO BWR environments, the CGRs of the irradiated steels decreased by an order of magnitude. The beneficial effect of decreased DO was not observed for Type 304 SS irradiated to 2 x 10^{21} n/cm^2. Type 304 SS irradiated to 0.3 x 10^{21} n/cm^2 (0.45 dpa) showed very little environmental enhancement of CGRs in the NWC BWR environment. The CGRs for Type 304 irradiated to 0.3 x 10^{21} n/cm^2 under SCC conditions are below the disposition curve for sensitized SSs in low-DO water (8 ppm), as given in NUREG–0313.

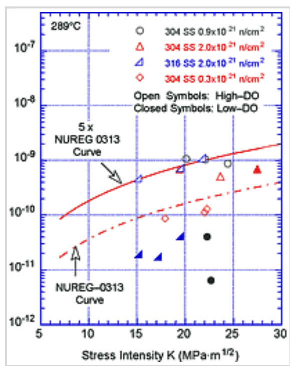


Figure b: SCC growth data for irradiated austenitic SSs in high-purity water at 289°C. Click on image to view larger image.

IASCC of Austenitic Stainless Steels in BWRs

Slow-strain-rate tensile (SSRT) tests in simulated BWR water were conducted on model austenitic SSs that were irradiated at 289°C in helium to ≈0.3 x 10^{21} n•cm^{-2} (≈0.4 dpa) and ≈2.0 x 10^{21} n•cm^{-2} (≈3 dpa) (E>1 MeV) in the Halden reactor. Fractographic analysis by scanning electron microscopy was conducted to determine susceptibility to IASCC as manifested by the degree of intergranular fracture surface morphology. Susceptibility to IASCC increased drastically as the damage level increased from ≈0.4 to ≈3 dpa. Bulk S content provided the best and the only good correlation with susceptibility to IASCC at >2 dpa. Type 304 and 316 SSs containing S ≤0.002 wt.% were resistant, whereas at S concentrations >0.003 wt.%, susceptibility increased drastically. This finding indicates that the deleterious effect of S is dominant at high fluence, and that an S-related critical phenomenon occurs. Sulfur content of ≤0.002

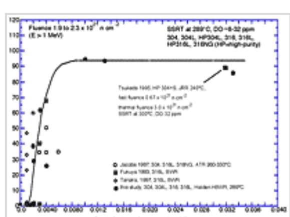


Figure c: Effects of S on susceptibility to intergranular SCC of Type 304 and 316 SSs (C content ≈0.060 wt.%) irradiated to 3 dpa. Click on image to view larger image.

wt.% does not necessarily render low-carbon steel (Types 304L, 316L) or high-purity-grade steels resistant to IASCC, indicating a beneficial effect of C.

Our two-dimensional map of bulk C and S concentrations shows the range in which Type 304, 304L, 316, and 316L SSs are resistant or susceptible to IASCC. This map is consistent with all data obtained for neutron-irradiated steels tested in BWR-like oxidizing water. Grain-boundary segregation of S was determined by Auger electron spectroscopy for BWR neutron absorber tubes fabricated from two Type 304 SS heats.

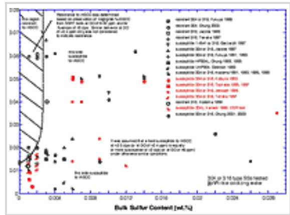


Figure d: Range of bulk S and C contents that renders a 304- or 316-Type steel susceptible or resistant to IASCC in BWR-like oxidizing water. Click on image to view larger image.

IASCC of Austenitic Stainless Steels in PWR Environment

Similar to the effort to evaluate the compositional characteristics of SS heats resistant to IASCC under BWR conditions, steels reported to be resistant to IASCC under PWR conditions were analyzed. Most of the data reported for PWR conditions were for Type 348 SSs. The two-dimensional maps of bulk S and C concentration (see figures) show the range in which Type 304, 316, and 348 SSs have been reported to be resistant to IASCC under PWR conditions. Compared with BWR oxidizing condition, good resistance to IASCC under PWR conditions is observed for a wider range of S concentration.

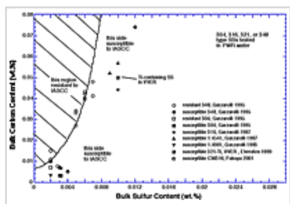


Figure e: Range of bulk S and C contents that renders austenitic steels susceptible or resistant to IASCC in PWR condition. Click on image to view larger image.

Susceptibility to intergranular fracture in 23°C air was determined for ≈3-dpa steels with and without exposure to the SSRT conditions in 289°C water. Similar bending tests were also performed on hydrogen-charged BWR components in 23°C vacuum. Both types of bend fracture exhibited similar characteristics dominated by hydrogen-induced irradiation-assisted cracking (IAC). Steels that showed high susceptibility to IASCC in 289°C water exhibited low susceptibility to hydrogen-induced IAC in 23°C air or vacuum, and vice versa. A heat that contains an unusually high content of S (0.022 wt.%) and an unusually low content of Mn (0.36 wt.%) failed as a result of virtually complete intergranular corrosion in 23°C air. This failure, not influenced by hydrogen, could not be explained on the basis of hydrogen-induced IAC but can be explained by a grain-boundary process in which Ni and S segregation plays an important role.

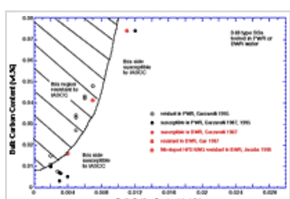


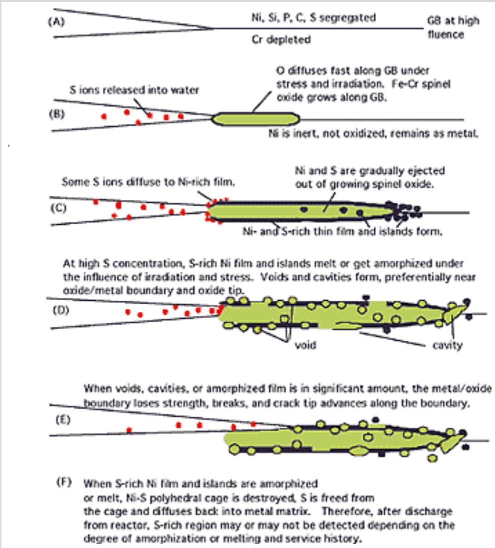
Figure f: Range similar to the one shown in figure e) applicable to Nb-stabilized steels for BWR and PWR conditions. Click on image to view larger image.

A large number of cold-worked and solution-annealed heats of Type 348, 316, and 304 SSs were irradiated to ≈5 and ≈10 dpa at 323°C in the BOR-60 reactor; irradiation to ≈40 dpa is in progress. The lower-dpa specimens are being transported from Russia for tensile and SCC testing in PWR water.

Initial Model of IASCC of Austenitic Stainless Steels

An initial model of IASCC was developed in consideration of several key observations in this study and elsewhere. The model is based on the following observations: (a)

dominant effect of S, (b) evidence of grain-boundary (GB) segregation of S in unirradiated steel and in irradiated core internal components, (c) properties of the Ni-S binary that contains S-segregated grain boundaries, and (d) crack-tip microstructure and microchemistry of irradiated steels. The key aspects of the model are illustrated in figure below and in the document:
Model of irradiation-assisted SSC in austenitic stainless steels [299KB].



Model of irradiation-assisted SSC in austenitic stainless steels. Download a larger printable version of the model in PDF format [299KB]

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