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Environmentally Assisted Cracking in Light Water Reactors Annual Report
January—December 2001, NUREG/CR–4667 Vol. 32, ANL–02/33 (June 2003).

Abstract

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) from January to December 2002. Topics that have been investigated include: (a) environmental effects on fatigue crack initiation in carbon and low–alloy steels and austenitic stainless steels (SSs), (b) irradiation–assisted stress corrosion cracking (IASCC) of austenitic SSs in BWRs, (c) evaluation of causes and mechanisms of irradiation-assisted cracking of austenitic SS in PWRs, and (d) cracking in Ni–alloys and welds.

A critical review of the ASME Code fatigue design margins and an assessment of the conservatism in the current choice of design margins are presented. The existing fatigue ε–N data have been evaluated to define the effects of key material, loading, and environmental parameters on the fatigue lives of carbon and low–alloy steels and austenitic SSs. Experimental data are presented on the effects of surface roughness on fatigue crack initiation in these materials in air and LWR environments.

Crack growth tests were performed in BWR environments on SSs irradiated to 0.9 and 2.0×10^{21} $n \cdot cm^{-2}$. The crack growth rates (CGRs) of the irradiated steels are a factor of ≈5 higher than the disposition curve proposed in NUREG–0313 for thermally sensitized materials. The CGRs decreased by an order of magnitude in low–dissolved oxygen (DO) environments.

Slow-strain-rate tensile (SSRT) tests were conducted in high-purity 289°C water on steels irradiated to ≈3 dpa. The bulk S content correlated well with the susceptibility to intergranular SCC in 289°C water. The IASCC susceptibility of SSs that contain >0.003 wt.% S increased drastically. Bend tests in inert environments at 23°C were conducted on broken pieces of SSRT specimens and on unirradiated specimens of the same materials after hydrogen charging. The results of the tests and a review of other data in the literature indicate that IASCC in 289°C water is dominated by a crack-tip grain-boundary process that involves S. An initial IASCC model has been proposed.

A crack growth test was completed on mill annealed Alloy 600 in high–purity water at 289°C and 320°C under various environmental and loading conditions. The results from this test are compared with data obtained earlier on several other heats of Alloy 600.

Foreword

For over 34 years Argonne National Laboratory has been a prime contractor to the Office of Nuclear Regulatory Research (RES) for studies of the environmental degradation of structural materials in light water reactor environments. This document is the 2002 annual report of the program studies. The program has evolved to keep pace with the most critical of the contemporary issues facing the industry and the NRC. Task 1 focused on the environmental degradation of fatigue life of pressure boundary materials. Task 2 addresses irradiation-assisted stress corrosion cracking (IASCC) of stainless steels in BWR environments, and a parallel program (Task 3) is addressing IASCC of stainless steels in pressurized water reactor (PWR) environments. Task 4, the study of crack growth rates in nickel-base alloys typically used in vessel penetrations, is currently focused on testing alloy 600 and its associated weld metal, alloys 182 and 82. Task 4 will test alloy 690 and its associated weld metal, alloys 152 and 52, which are the materials of choice for most replacement vessel head penetrations.

In earlier years, ANL research produced the finding that the fatigue life of stainless steels is degraded to a greater degree in de-oxygenated, PWR-like environments than it is in boiling water reactor (BWR) environments. Research completed in 2002 provided added confirmation of that result, further characterized the microstructural aspects of fatigue crack initiation, and evaluated the effects of surface roughness on fatigue life degradation of low-carbon steels, low-alloy steels, and stainless steels. The database for the environmental degradation of fatigue lives in stainless steels buttresses the NRC position *vis a vis* the ASME code - that the underlying computational logic and application of the design curves for the fatigue life of pressure boundary and internal components fabricated from stainless steel is non-conservative, and needs revision.

Cracking of nickel-base alloys commonly used in vessel penetrations was initially manifested in pressurizer nozzles and heater sleeves, which normally operate in a temperature range somewhat higher than other reactor components . When cracking was observed in vessel head penetrations, RES incorporated crack growth rate studies in the ANL test program. Although most plant observations of cracking have occurred in PWRs, the ANL test program is testing these materials under both BWR and PWR conditions. These results will be used to support flaw evaluations and the associated requests for continued operations that are proposed to the NRC.

In the future, the IASCC work will test materials that have received higher radiation doses, and will involve more microstructural characterization of such materials. Studies of void swelling and stress corrosion cracking studies of cast or welded stainless steels will also enter the test program. The SCC studies of nickel-base alloys will begin to refocus on alloys 690 and its associated weld metal, alloy 152, including cold-worked and heat-affected zone forms of the wrought material.

Carl J. Paperiello, Director
Office of Nuclear Regulatory Research

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Executive Summary

The existing fatigue ε–N data for carbon and low–alloy steels and wrought and cast austenitic SSs have been evaluated to define the effects of key material, loading, and environmental parameters on the fatigue lives of these steels. The fatigue lives of carbon and low–alloy steels and austenitic SSs are decreased in LWR environments. The magnitude of the reduction depends on temperature, strain rate, DO level in water, and, for carbon and low–alloy steels, S content in steel. The threshold values of the critical parameters and the effects of other parameters (such as water conductivity, water flow rate, and material heat treatment) on the fatigue life of the steels are summarized.

Experimental data are presented on the effects of surface roughness on the fatigue life of carbon and low–alloy steels and austenitic stainless steels in air and LWR environments. For austenitic SSs, the fatigue life of roughened specimens is a factor of ≈3 lower than that of the smooth specimens in both air and low–DO water. For carbon and low–alloy steels, the fatigue life of roughened specimens is lower than that of smooth specimens in air but is the same in high–DO water.

Statistical models are presented for estimating the fatigue life of carbon and low–alloy steels and wrought and cast austenitic SSs as a function of material, loading, and environmental parameters. Two approaches are presented for incorporating the effects of LWR environments into ASME Section III fatigue evaluations.

Because of material variability, data scatter, and component size and surface, the fatigue life of actual components is different from that of laboratory test specimens under a similar loading history, the mean ε–N curves for laboratory test specimens are adjusted by factors of 2 on stress and 20 on cycles to obtain design curves for components. These factors should not be considered safety margins, but they were intended to cover the effects of variables that can influence fatigue life but were not investigated in the tests that provided the data for the curves. Data available in the literature have been reviewed to evaluate the margins on cycles and stress. The results indicate that the current ASME Code requirements of a factor of 2 on stress and 20 on cycle are reasonable, and do not contain excess conservatism that can be assumed to account for the effects of LWR environments.

Crack growth tests have been performed in simulated BWR environments at ≈289°C on Type 304 SS (Heat C3) irradiated to 0.9 and 2.0×10^{21} $n \cdot cm^{-2}$ and Type 316 SS (Heat C16) irradiated to 2.0×10^{21} $n \cdot cm^{-2}$ at ≈288°C in a helium environment. The results indicate significant enhancement of CGRs of irradiated steel in the normal water chemistry BWR environment. The CGRs of irradiated steels are a factor of ≈5 higher than the disposition curve proposed in NUREG–0313 for sensitized austenitic SSs in water with 8 ppm DO. Actual enhancement in the same purity water is greater than 5. The CGRs of Type 304 SS irradiated to 0.9 and 2.0×10^{21} $n \cdot cm^{-2}$ and of Types 304 and 316 SS irradiated to 2.0×10^{21} $n \cdot cm^{-2}$, are comparable.

In low–DO environment with low electrochemical potentials (ECPs), the CGRs of the irradiated steels decreased by an order of magnitude in tests in which the K validity criterion was satisfied, e.g., Heat C3 of Type 304 SS irradiated to 0.9×10^{21} $n \cdot cm^{-2}$ and Heat C16 of Type 316 SS irradiated to 2×10^{21} $n \cdot cm^{-2}$. No beneficial effect of decreased DO was observed for Heat C3 of Type 304 SS irradiated to 2×10^{21} $n \cdot cm^{-2}$, but in this case the applied K values during the low ECP portion of the test exceeded those required to meet the K validity criterion.

Slow-strain-rate tensile (SSRT) tests were conducted in high-purity 289°C water on steels irradiated to ≈3 dpa in helium in the Halden Reactor. At ≈3 dpa, the bulk S content provided the best and the only good correlation with the susceptibility to intergranular (IG) SCC in 289°C water. Good resistance to IASCC was observed in Type 304 and 316 stainless steels that contain very low concentrations of S of ≈0.002 wt.% or less. The IASCC susceptibility of Type 304, 304L, 316, and 316L steels that contain >0.003 wt.% S increased drastically. Steels containing ≥0.008 wt.% were very susceptible at high fluence. These observations indicate that the deleterious effect of S plays a dominant role in the failure of core internal components at high fluence.

In contrast to Type 304 and 316 stainless steels, a low concentration of S of ≈0.001-0.002 wt.% does not necessarily render low-carbon Types 304L and 316L, or high-purity-grade steel resistant to IASCC. This suggests that high concentration of C is beneficial in reducing the deleterious effect of S and that threshold S concentration to ensure good IASCC resistance is lower in a low-carbon steel than in a high-carbon steel.

A comprehensive irradiation experiment in the BOR-60 Reactor is under progress to obtain a large number of tensile and disk specimens irradiated under PWR-like conditions at ≈325°C to 5, 10, and 40 dpa. Irradiation to ≈5 and ≈10 dpa has been completed.

Tests performed on the materials irradiated in the Halden BWR reactor may also give some insight into potential mechanisms for IASCC that are also relevant to PWRs. After exposure to the conditions of the SSRT test in BWR water, susceptibility to intergranular cracking in an inert environment was determined by rapid bending in air at 23°C. Similar tests were also performed on hydrogen-charged specimens in vacuum. Both types of bend fracture exhibited similar characteristics suggesting that in both cases the failures occurred due to hydrogen-induced intergranular failure. However, steels that showed high susceptibility to IGSCC in 289°C water exhibited low susceptibility to intergranular cracking in the tests at 23°C air or vacuum, and vice versa. This indicates that although intergranular cracking in 23°C is dominated by H-induced embrittlement of ordinary grain boundaries, other processes control IASCC in 289°C water. On the basis of this investigation, and studies on binary Ni–S and crack-tip microstructural characteristics of LWR core internal components reported in literature, an initial IASCC model has been proposed.

The resistance of Ni alloys to environmentally assisted cracking in simulated LWR environments is being evaluated. A crack growth test was completed on mill annealed (MA) Alloy 600 (Heat NX131031) specimen in high–purity water at 289 and 320°C under various environmental and loading conditions. The results from this test are compared with data obtained earlier on several other heats of Alloy 600.

In a high–DO environment at 289°C, nearly all of the heats and heat treatment conditions that have been investigated show enhanced growth rates. The growth rates for MA (Heat NX131031) are slightly higher than for the other heats of Alloy 600. In contrast to the behavior in high–DO water, environmental enhancement of fatigue CGRs of Alloy 600 in low–DO water seems to depend on material condition, e.g., materials with high yield strength and/or low grain boundary coverage of carbides.

The SCC crack growth rates of Heat NX131031 in high–DO water at 289°C are comparable to those in low–DO water at 320°C. The results from the present study are compared with data obtained on several other heats of Alloy 600. In a PWR environment, the CGR of Heat NX131031 corresponds to the 53rd percentile of the distribution for the sample of heats of Alloy 600. For example, Heat NX131031 represents an average heat.

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