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Statistical Analysis and Life Prediction on IASCC of Core Internals in a PWR

Sung-Woo Kim *, Seong-Sik Hwang
Nuclear Materials Division, Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Daejeon 305-353
**Corresponding author: kimsw@kaeri.re.kr*

1. Introduction

Irradiation assisted stress corrosion cracking (IASCC) has been regarded as the main cause for intergranular cracking incidents in core internals in light water reactors (LWRs). IASCC was reported in a fuel rod in the 1960s, a control rod in the 1970s, and a baffle former bolt in recent years. For a proactive management of IASCC of these components, a lot of work has been performed in boiling water reactors (BWRs) [1, 2]. From these works, radiation-induced segregation (RIS), neutron fluence, and applied stress were reported to be crucial factors affecting the IASCC susceptibility of stainless steel. Recently, many efforts have also been to investigate the IASCC of the core internals in pressurized water reactors (PWRs), but the mechanism in PWRs is not fully understood yet as compared with that in BWRs owing to a lack of reliable data. Therefore, this work is concerned with a statistical analysis of factors affecting an IASCC failure, and the life prediction of austenitic stainless steels under a complex environment of severe neutron irradiation, and high temperature and pressure in PWRs, for establishing a proactive management program of Korean PWRs.

2. Statistical analysis and modeling

A statistical evaluation of the experimental data was performed using MINITAB^{TB} software (release 13.20). The relationship between the SCC susceptibility (%IGSCC) and Cr depletion at the grain boundary (GB) of irradiated stainless steel was evaluated using a correlation analysis tool. The effect of neutron fluence on the SCC susceptibility was investigated by a probit analysis, and the threshold fluence of IASCC was statistically estimated at various temperatures. In addition, the effects of the applied stress and irradiation fluence on the time-to-failure of IASCC were studied based on a newly established accelerated life testing (ALT) model [3].

$$\ln[L(\sigma, d)] = \ln(A) - m\ln(\sigma) - n\ln(d) + p\Phi^{-1}(q) \qquad (1)$$

where L is life time, σ is the ratio of yield strength of irradiated material to non-irradiated material (%), d is the neutron fluence in DPA, $\Phi^{-1}(q)$ is the qth percentile of the standard normal distribution, p is the scale parameter of a Weibull distribution, and A, m, and n are constants.

3. Results and discussion

Fig.1 shows the susceptibility of SCC as a function of the degree of Cr depletion at the GB owing to irradiation collected from the previous work [3]. All data were measured from a slow strain rate test (SSRT) under the same experimental conditions; that is, in the normal water chemistry of a BWR, when the content of dissolved oxygen (DO) is about 32 ppm at 288 ^{o}C, and at a similar strain rate of 1×10^{-7}/s. %IGSCC increased with a decrease in Cr content at the GB.

From the experimental data in Fig. 1, the Pearson correlation coefficient and P-value were calculated using a correlation analysis. In the case of thermally sensitized stainless steel, the P-value was lower than the significance level, α of 0.05, indicating that the SCC susceptibility strongly correlated with Cr depletion at the GB. For irradiated stainless steel, however, the P-value was higher than α of 0.05. This means that the SCC susceptibility and Cr depletion at the GB did not correlate with each other; that is, Cr depletion at the GB was not the main factor of the irradiated stainless steel. Recently, much work on microstructural factors has pointed out that the RIS of minor elements such as Si, P, and S in BWRs [1, 2], and the Ni enrichment in PWRs [1] are other factors affecting IASCC.

Fig. 2 shows the SCC susceptibility of stainless steel irradiated to various neutron fluences, measured from the SSRT under the same conditions at the primary water chemistry of the PWR (DH ~ 30 cc/kg) at various temperatures. %IGSCC remarkably increased with an increase in fluence. The relation between %IGSCC and the neutron fluence was found to follow a Weibull distribution. From a probit analysis, the threshold fluence of IASCC of stainless steel was estimated as the percentile at a 2% failure probability. It should be noted that the threshold value decreased remarkably from 5.799 to 5.253 and 1.914 DPA with an increase in temperature from 320 to 325 and 340 ^{o}C, respectively.

Fig. 3 shows the time-to-failure of irradiated stainless steel at various neutron fluences in the PWR, measured from a constant elongation test under the same conditions of the primary water chemistry of the PWR (DH ~ 30 cc/kg and 340 ^{o}C). From a distribution analysis, it was confirmed that the experimental data follows the Weibull distribution fairly well, and both applied stress and neutron fluence were independent accelerants for IASCC (Fig. 4(a) and (b)). Based on the ALT model, the remaining life time of a baffle-former bolt (BFB), one of the significant core internals in PWRs, under complex environments of irradiation and stress was statistically estimated (Fig. 5) under five different operating and repair conditions: (case 1) BFB operated without repair, (case 2) BFB repaired with a 100% preload after 10 years of operation, (case 3) BFB repaired with a 100% preload after 40 years of operation, (case 4) BFB repaired with a 90% preload after 10 years of operation, (case 5) BFB repaired with a 70% preload after 40 years of operation.

4. Conclusion

The Cr depletion at the GB was determined to have no significant correlation with the IASCC susceptibility. The threshold irradiation fluence of IASCC in a PWR was statistically calculated to decrease from 5.799 to 1.914 DPA with an increase in temperature from 320 to 340 ^{o}C. From the analysis of the relationship between applied stress and the time-to-failure of stainless steel components, the B2 life time of the irradiated baffle former bolt was successfully estimated based on the newly established ALT model.

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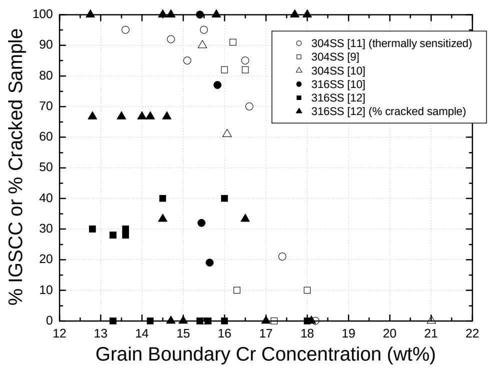

Fig. 1. Comparison between the Cr concentration at the GB and %IGSCC of thermally sensitized and neutron-irradiated stainless steel in a BWR normal water chemistry.

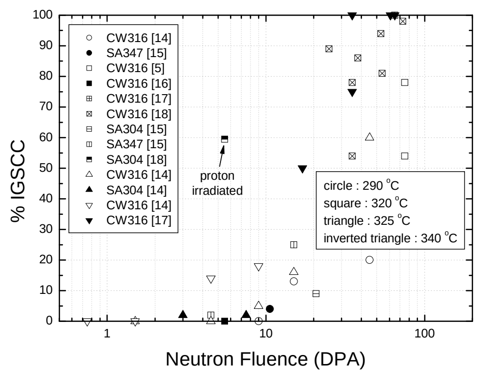

Fig. 2. %IGSCC of various stainless steel at various neutron fluence in a PWR primary water chemistry.

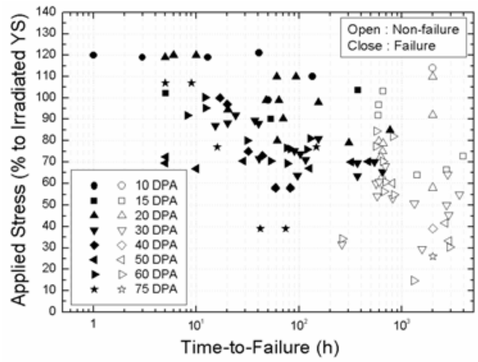

Fig. 3. The relation between %IGSCC and the neutron fluence in a PWR primary water chemistry.

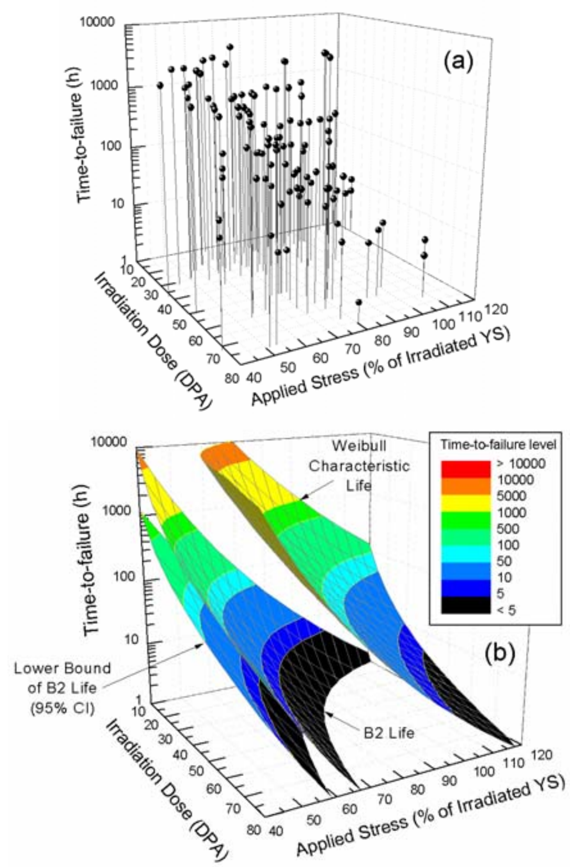

Fig. 4. 3D plot of time-to-failure of neutron-irradiated CW 316SS (a) measured and (b) estimated from the ALT model.

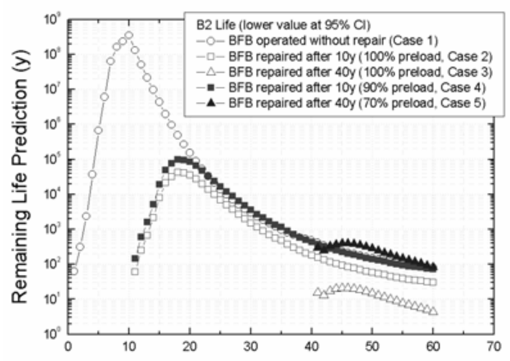

Fig. 5. Lower value of remaining B2 life time of neutron-irradiated CW 316SS estimated at 95% CI under various operating and repair conditions.