Review on IASCC initiation of 316 stainless steels in a PWR environment

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1. Introduction

Irradiation assisted stress corrosion cracking (IASCC) involves the cracking and failure of materials in nuclear power plant water environment. The major factors and processes governing an IASCC are suggested in Fig. 1 [1].

The IASCC of the reactor core internals due to the material degradation and the water chemistry change has been reported in high stress stainless steel components, such as fuel elements (Boiling Water Reactors) in the 1960s[1], a control rod in the 1970s, and a baffle former bolt in recent years of light water reactors (Pressurized Water Reactors).

Under the circumstances, a lot works have been performed on IASCC in BWR. Recent efforts have been devoted to investigate an IASCC in a PWR, but the mechanism in a PWR is not fully understood yet as compared with that in a BWR owing to a lack of data from laboratories and fields. Therefore, it is strongly necessary to review and analyze recent researches of an IASCC in both BWR and PWR for establishing a proactive management technology for the IASCC of core internals in Korean PWRs [2].

The objective of this research to find IASCC behavior of stainless steels in a high-temperature water chemistry environment with un-irradiated stainless steel specimens and proton irradiated stainless steels.

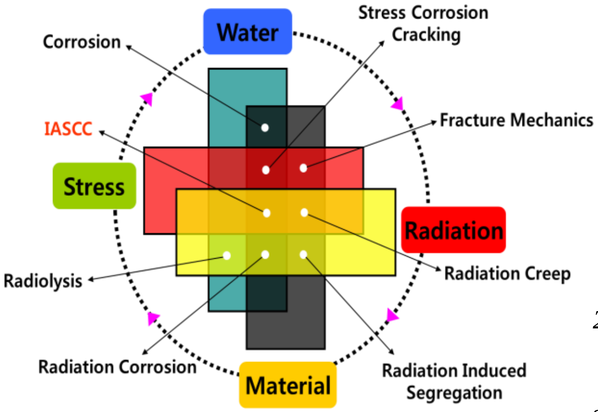

Fig. 1. Schematic diagram of factors to IASCC [1]

2. Experimental Procedures

2.1 Materials

As-received 316 stainless steels in a PWR environment are examined in this study. Table 1 shows the chemical compositions of the tested alloy.

Specimens were fabricated with a gage length of 23 mm by electric discharge machining. A schematic of the sample is shown in Fig. 2. The surfaces of the specimens were mechanically wet-polished using #400-#2400 SiC sand paper, and then electro-polished for 15-30 seconds, in a 50% phosphoric acid, 25% sulfuric acid, and 25 % Glycerol at room temperature.

Table 1 Chemical compositions of tested alloy, wt %

Fe	Bal.
Ni	10.8
Cr	16.7
Mo	2.0
Mn	1.3
Si	0.59
P	0.05
C	0.047
S	0.001

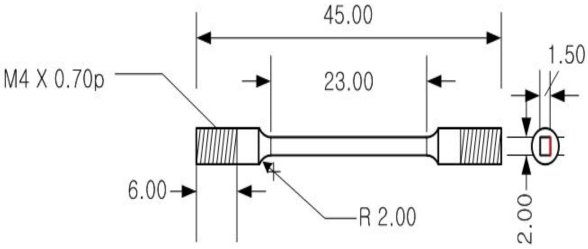

Fig. 2. Schematic of tensile specimen

2.2 Stress corrosion cracking experiments

Slow strain rate tests (SSRT) on un-irradiated samples were conducted in a multiple-specimen SSRT test system, which is able to strain four samples in parallel, providing equal conditions within a given test [3].

The conditions of the SSRT tests were 1200 ppm B - 2 ppm Li chemistry conditions of PWR primary water. Also, the SSRT setup consisted of an autoclave capable of sustaining pressures up to 2400 psi and temperatures up to 340 ^oC. The water chemistry was characterized by a water pH 6.35, conductivity of 21.12 μS/cm, and dissolved oxygen of 5 ppb, and dissolved hydrogen of 25 cc/kg. Multi-specimens were strained to failure at a rate of 3.0×10^{-7}/s. The SSRT tests conditions are described in Table 2.

Table 2 SSRT test conditions

Parameter	PWR primary water
Strain rate	3.0×10^{-7}/s
Temperature	340 ^oC
pH	6.35
Conductivity	21.12 μS/cm
Dissolved Oxygen	<5 ppb
Dissolved Hydrogen	~25 cc/kg

3. Summary

IASCC tests were accomplished for un-irradiated stainless steel specimens in PWR simulated condition.

The IASCC tests with 1, 3, 5, 10 dpa samples proton irradiated will be conducted near future.

IASCC tests results will be evaluated in terms of obtained stress-strain curves, a surface fracture SEM image, the crack density, crack length, and crack length per unit area.

REFERENCES

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[2] S.W. Kim et al., Characterization of Factors affecting IASCC of PWR Core Internals, KAERI/AR-811/2008

[3] J.T. Busby and G.S. Was, Effect of Metallurgical Condition on Irradiation-Assisted Stress Corrosion Cracking of Commercial Stainless Steels, Journal of Nuclear Materials 300 (2002) 198-216