A Review on Stress Corrosion Cracking of Stainless Steel 316L in Oxygenated and Chlorinated Primary Water Chemistry

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1. Introduction

STS 316L austenitic stainless steel has been used widely as structural material under primary water reactor environment in PWR due to high resistance of general corrosion and mechanical strength. However, there are many localized corrosion failures in nuclear reactor, and stress corrosion cracking (SCC) is occurred. SCC is classified as either intergranular stress corrosion cracking (IGSCC) or transgranular stress corrosion cracking (TGSCC), depending upon the primary crack morphology. Since 1980s, stress corrosion cracking became an important degradation mechanism to deteriorate reliability of components in nuclear power plants. SCC has been studied extensively over the last thirty years, the cracking process is still a matter of debate [1, 2].

To prevent SCC, it is necessary to know boundary condition that SCC occurs. In this study, previous studies of boundary condition generating SCC related to austenitic STS 316L in water chemistry are analyzed.

2. Methods and Results

2.1 Factors affecting SCC

Stress corrosion cracking is the macroscopic brittle failure of ductile material through slow environment change induced crack propagation. Commonly, SCC is occurred by interaction among material, tensile stress and corrosive environment. Prior studies indicated that the susceptibility of SCC is related to the characteristic of oxide material which is dependent on water-chemistry conditions [3].

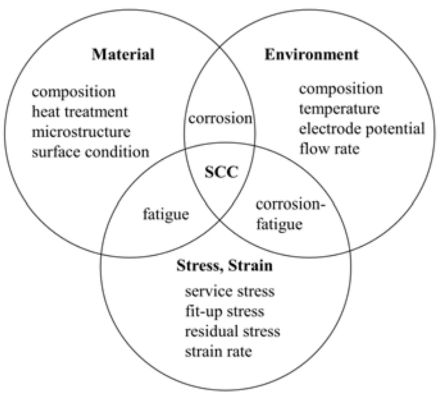

Fig. 1. Factors affecting stress corrosion cracking [4]

2.2 Slow strain rate test (SSRT)

When tensile specimen is exposed to the corrosive environment, the test (such as constant strain test, constant load test and slow strain rate test) obtain the desired data with very slow speed and constant elongation rate.

Slow strain rate test (SSRT) provides a rapid and reliable method to determine susceptibility of SCC for metals and alloys. The advantage of SSRT is to produce SCC faster than conventional constant strain or constant load tests, so test time is considerably reduced.

The typical stress-strain test uses strain rate of approximately 10^{-2} s^{-1}, so it takes a few minutes. On the other hand, SSRT uses strain rate of 10^{-6} s^{-1}, so it takes at least two days.

Previous study discussed the effect of the strain rate on the stress corrosion cracking of STS 316 austenitic stainless steel in simulated PWR water at 325℃. The stress-strain curve at different strain rate (2×10^{-7} s^{-1} and 2×10^{-8} s^{-1}) shows the shorter elongation and the lower maximum stress at 2×10^{-8} s^{-1}. This result indicated that it is more sensitive for SCC at lower strain rate [5].

2.3 Water chemistry conditions

Water chemistry of primary condition affected by radiation exposure at PWR is controlled during operation of nuclear power plant. Important factor of this condition includes lithium hydroxide to control pH, boric acid to help reactivity of core and sufficient dissolved oxygen and hydrogen to suppress decomposition of water by radiolysis [6].

According to previous study, the experiment of SCC susceptibility at 150℃ in purity water with various dissolved oxygen concentration (DO < 0.05 ppm, DO 0.3 ~ 0.4 ppb, DO 8 ppm) was conducted by SSRT. At the highest DO level (8 ppm), brittle fracture occurs. Fracture surfaces show that intergranular stress corrosion cracking (IGSCC) is occurred along the grain boundary. In comparison to experiment in high-purity water condition, dissolved oxygen deteriorates mechanical property of specimen. The sensitivity of SCC increased with elevation of dissolved oxygen concentration. It is occurred by the oxidation and the reduction reaction in specimen [7]. Other experiment investigated the IGSCC susceptibility of metals (STS 304 and STS 316) having different carbon contents in dissolved oxygen (8 ppm) and chloride (Cl < 0.05 ppm) water. Both deionized high-purity water and borated water (2,100 ppm as B) were used at 175 and 240°C [8]. SSRT was conducted at 200℃ in primary water condition (DO 3 ppm, Cl 11.1 ppm). The environment was simulated PWR primary water (1000 ppm B or 1200 ppm B added as boric acid, 2 ppm Li added as lithium hydroxide). Prior to this test, specimens had a pre-oxidation period of 1000h for reducing required DO for cracking. The cracks were essentially transgranular, although some intergranular cracks were also investigated [9].

SSRT was performed on STS 316 in 265 °C water containing oxygen (from 0 to 45 ppm) and chloride (from < 0.1 to 1000 ppm). Congleton [10] further extended the cracking regions with various levels of oxygen and chloride. But it didn't involve addition of primary water condition such as lithium, boric acid, or hydrogen.

Table I. SCC data in various oxygen and chloride [7-10]

O_2 (ppm)	Cl (ppm)	Type of cracking*	Temp. (°C)
8	0	IG	150
8	< 0.05	IG	175
3	11.1	IG,TG	200
0	45	TG	265
< 0.1	0	TG	
< 0.1	0.55	TG	
2	45	TG	
10	45	TG	
10	9	TG	
0	45	IG,TG	
0	9	IG	
< 0.1	1.1	IG	
< 0.1	0.58	IG	
2	45	IG	
2	9	IG	
2	0.2	IG	
2	0.02	TG	
10	45	IG	
10	9	IG	

* IG : Intergranular stress corrosion cracking
TG : Transgranular stress corrosion cracking

As shown in table 1, SCC is occurred by various conditions of oxygen and chloride. It shows that dissolved oxygen affected intergranular stress corrosion cracking and increased chlorine level tended to transgranular stress corrosion cracking with the exception of some cases. Figure 2 shows that a few amount of chemical water (such as oxygen and chloride) can cause SCC. Depending on the condition boundary of SCC with the minimum values, it helps to guess if there is potential SCC occurrence. However, the environment (such as temperature, pressure, material, pH, etc) is different for each study and these differences may result in different consequences. Therefore, many experimental data needed under the same conditions.

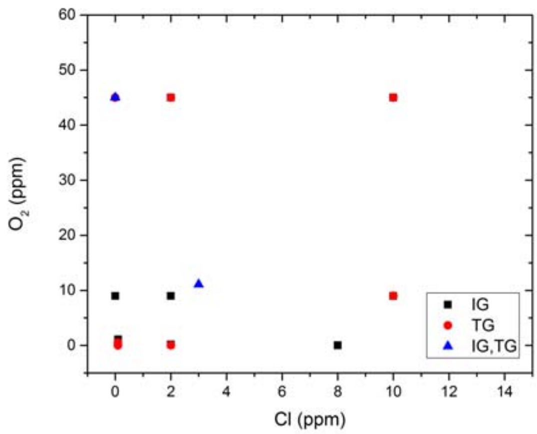

Fig. 2. The effects of oxygen and chloride on SCC [7-10]

3. Conclusions

The stress corrosion cracking (SCC) is a critical for ensuring the integrity of the nuclear power plant.

In this paper, a summary of SCC boundary condition through results of studies on the SCC simulated experiment is presented. But the findings in the same environment are insufficient to satisfy the SCC boundary. Therefore, experiments should be conducted to find correct boundary condition of SCC under same conditions in the future.

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