Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 22, 2009

Stress corrosion cracking of Ni base alloys in Pb-contaminated caustic solutions

Mi-Ae Kim [*], Dong Jin Kim, Joung Soo Kim, Hong Pyo Kim
Nuclear Materials Research Division, Korea Atomic Energy Research Institute(KAERI),
Yuseong-gu, Daejeon, 305-353
[*]*Corresponding author: makim@kaeri.re.kr*

1. Introduction

Steam generator tubes in pressurized water reactors(PWRs) form a pressure boundary between the primary and secondary sides. Austenitic stainless steel was used initially for the tubing, but it was changed to Alloy 600 as the results of corrosion problems. Experiences with Alloy 600 in the late 1960s and early 1970s, in the early high temperature large PWRs, identified numerous corrosion problems. Efforts for the development of a new made and the result was Alloy 690. Since the mid-1980s new and replacement steam generators have used Alloy 690. However, PWRs with Alloy 600 steam generators are still being operated.

A lot of problems related to corrosion have been reported in Alloy 600 steam generator tubes of operating nuclear power plants(NPPs), and the outer diameter stress corrosion cracking (ODSCC) and intergranular attack(IGA) which have been occurring in Alloy 600 tubes are known to be the leading causes of PWR steam generator tube plugging in the USA and worldwide [1]. According to Smith and Stratton, Alloy 690 is also reported to be susceptible to stress corrosion cracking and intergranular attack in deaerated (with argon) caustic solutions [2].

It has been reported that contaminated lead in the secondary side cooling water is accumulated in the sludge piled on top of the tube sheet thus accelerating stress corrosion cracking in the SG tubes of NPPs. The detailed mechanism of accelerated stress corrosion cracking of Alloy 600 and 690 by Pb, however, has yet to be completely understood. It was observed that Pb dissolved in water can produce PbSCC at a Pb concentration as low as 0.1 ppm in these alloys [3]. Lead is known to be one of the most aggressive environmental species that can accumulate in the crevice between the tubes and sludge piled on TTS in steam generators. Many laboratory experiments indicate that the stress corrosion cracking of steam generator tubing materials is accelerated in the presence of lead species in a caustic environment [4-5].

In order to observe the effect of Pb on stress corrosion cracking, it is needed to investigate stress corrosion cracking susceptibility of Alloy 600 and 690 in solutions with Pb.

In this study, stress corrosion cracking tests were carried out with Alloy 600 and 690 in caustic solutions with Pb.

2. Experimental

The materials used in this study were Alloy 600 HTMA and 690 TT. The chemical compositions are given in Table 1 and 2.

Slow strain rate tensile(SSRT) tests were performed in caustic solutions containing PbO at 315℃. The SSRT test is known to take a relatively shorter time and show better reproducibility compared to other test methods. The SSRT tests were carried out using a Ni autoclave of 0.5 gallon at 315℃ and an open circuit potential.

Figure 1 shows the shape and dimensions of the SSRT test samples used in this study. The strain rate of the SSRT test was $2 \times 10^{-7} s^{-1}$.

Surfaces in the gauge section and the fracture surfaces after the SSRT tests were examined with a scanning electron microscope (SEM). From the SEM observations, the crack morphology and the SCC areas were determined.

3. Results and discussion

Figure 2 shows the stress-strain curves of the Alloy 690 TT tested in 0.1M NaOH containing PbO with /without deaeration. As shown in this figure, the elongation and ultimate tensile strength (UTS) of Alloy 690 TT tested in 0.1M NaOH containing PbO with deaeration was approximately 55% and 597MPa, respectively, whereas those in 0.1M NaOH containing PbO without deaeration was approximately 54% and 593MPa, respectively. There is no apparent difference in the elongation and UTS of the Alloy 690 TT.

Figure 3 shows SEM micrographs that illustrate the fracture surface and specimen surface morphologies of a SSRT specimen tested in 0.1M NaOH containing PbO without deaeration. The fracture surfaces of the specimen show a ductile fracture with dimples as shown in Figure 3. The specimen surface in the gauge section did not show significant SCC except for few cracks, as shown in Figure 3. The Alloy 690 TT shows superior effectiveness of SCC resistance in 0.1M NaOH containing PbO.

3. Conclusions

The stress corrosion cracking susceptibility of Alloy 690 TT was investigated in caustic solutions containing PbO. From the experimental observations, it can be concluded as follows;

1. In order to evaluate the effect of a deaeration of the tested solutions, SSRT tests were performed with the

Alloy 690 TT specimens in 0.1M NaOH containing PbO with or without a deaeration. From the SSRT tests, there is no significant difference in the elongation and UTS of the Alloy 690 TT.

2. The SEM micrographs show a superior effectiveness of the SCC resistance in 0.1M NaOH containing PbO.

REFERENCES

[1] D. R. Diercks, W. J. Shack, J. Muscara, Nucl. Eng. Des., Vol.194, p.19, 1999.

[2] A. J. Smith, R. P. Stratton, in: Proceedings ot the 5th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, Monterey, CA, 25-29 August, American Nuclear Society Inc., La Grange Park, IL, pp. 855-860, 1991.

[3] R. W. Staehle, Proc. Of 11th International Symposium on Environmental Degradation of Metals in Nuclear Power Systems – Water Reactors, Stevenson, Washington, P.381, 2003.

[4] M. Wright, M. Mirzai, Lead-induced SCC propagation rates in Alloy 600, in: Proceedings of Ninenth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, Newport Beach, California, p. 657, 1999.

[5] M. Psaila-Dombrowski, F. Hua, P. Doherty, Lead assisted stress corrosion cracking of alloy 690, in: Proceedings of Ninth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, Newport Beach, California, p. 703, 1999

Table 1 Chemical compositions of Alloy 600 HTMA (wt %)

Material	C	Si	Mn	P	Cr	Ni	Fe
Alloy600	0.025	0.05	0.22	0.07	15.67	75.21	8.24
Material	Co	Ti	Cu	Al	B	S	N
Alloy600	0.005	0.39	0.011	0.15	0.0014	0.001	0.0103

Table 2 Chemical compositions of Alloy 690 TT (wt %)

Material	C	Si	Mn	P	Cr	Ni
Alloy690	0.05 max	0.5 max	0.5 max	0.015 max	27~31	58
Material	Fe	Ti	Cu	Al	S	
Alloy690	7~11	0.04 max	0.5 max	0.4 max	0.015 max	

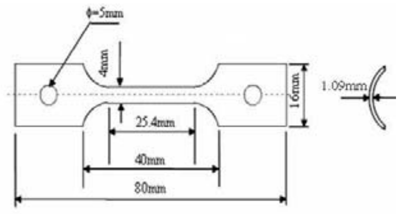

Figure 1. Shape and dimensions of the slow strain rate tensile specimen.

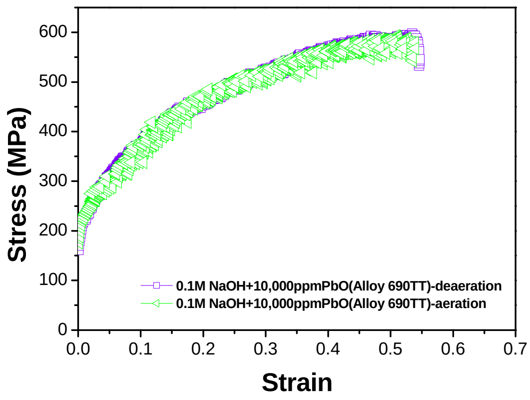

Figure 2. Stress-strain curves of Alloy 690 TT in 0.1M NaOH + 10,000ppm PbO with/without deaeration at 315℃.

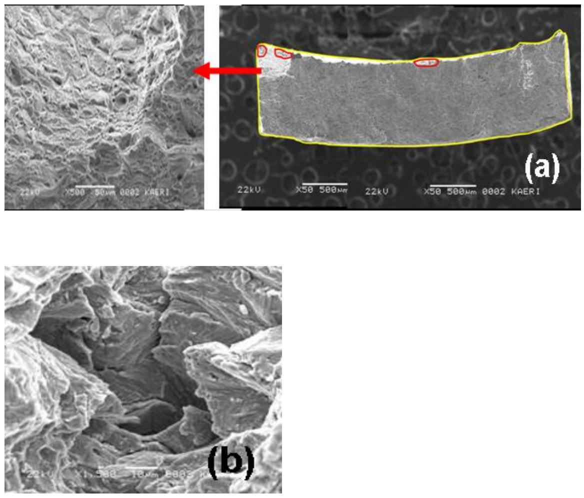

Figure 3. SEM micrograph of the fracture surface(a) and specimen surface morphologies(b) Alloy 690 TT in 0.1M NaOH+10,000ppm PbO without deaeration at 315℃.