Degradation analysis strategy on Alloy 690 SG tubings of a retired steam generator

Seong Sik Hwang *, Sung Hwan Cho, Sung Woo Kim , Dong Jin Kim
Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon 34057, Korea
**Corresponding author: sshwang@kaeri.re.kr*

1. Introduction

A study to analyze the degradation of components that have been exposed to high-temperature, high-pressure and neutron radiation by harvested out components of a nuclear power plant that was shut down in 2017 has recently started in Korea. The steam generator(SG), which has an operating history of 19 years until it was shut down in 2017, is made of alloy 690. By accurately analyzing the characteristics of the surface oxide of the SG tubings and comparing it with laboratory corrosion data, it will be possible to predict the degradation of the alloy 690 tubings due to long-term operation. From the viewpoint of material degradation analysis, it is necessary to examine whether there is evidence of long range ordering/short range ordering (LRO/SRO), especially in the material operated near at 320 ºC, and a material analysis method is needed for this.

Based on the above research background, in this paper, we would like to introduce the harvesting strategy of the decommissioned SG tubings and the method of research to be carried out by using this material.

2. Background

A study was recently started in Korea to analyze the degradation of components that have been exposed to high temperature, high pressure and neutron radiation by harvesting components from nuclear power plants that were decommissioned in 2017. This is a study to develop empirical evaluation technology for pressure boundary material degradation and establish an empirical evaluation system using materials from existing nuclear power plants.

Among them, the SG tubings are representative core components related to the safety of operating nuclear power plants, and the existing alloy 600 SG tubings showed defects such as primary water stress corrosion cracking (PWSCC) as shown in Fig. 1.

In some power plants in Korea, there was a case where power generation was stopped due to a leakage caused by stress corrosion cracking in the SG tubings, and axial cracking defects were confirmed in the tubings adjacent to the tube support plate.

There have been no reports of corrosion cracking cases worldwide for alloy 690 SG tubings so far, and corrosion resistance has been improved a lot compared to alloy 600. Therefore, it is necessary to evaluate the conservatism of laboratory data and to demonstrate the actual degree of degradation by acquiring material degradation data from the alloy 690 SG tubings.

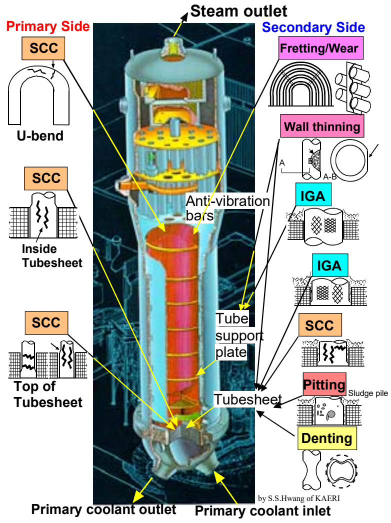

Fig. 1. Corrosion/cracking degradation in SG of nuclear power plants

In addition, by analyzing the causes of denting caused by long-term operation, it will be possible to use it to mitigate damage such as denting of tubings in subsequent nuclear power plants.

2.1 Operation history of the steam generator

The SG tubings of the power plant were alloy 600, which is vulnerable to stress corrosion cracking. After being replaced with a Westinghouse-type delta 60 using alloy 690 in 1998, it has a 19-year operating history until it was shut down in 2017.

This alloy 690 tubings can provide important information that can empirically evaluate the corrosion condition of the inner and outer surfaces of the tubings. By accurately analyzing the characteristics of the surface oxide, it will be possible to predict the degradation of alloy 690 due to long-term operation.

A tube plug made of alloy 690 was used for the SG of this decommissioned nuclear power plant. The 15 tubings were welded plugged before operation, and 8 tubings with wear defects were plugged by using a mechanical method based on ISI(In service inspection) performed in 2014. In the case of mechanical plugs of SG tubings, the plastic strain rate is up to 50%, so the PWSCC sensitivity of these materials is similar to alloy 600. Some US plants already showed some cracking in alloy 600 plugs.[1] Fig. 2 shows an example of PWSCC cracking in alloy 600 tube plugs. Therefore, it is necessary to take out a tube plugs made of alloy 690 and verify in advance by evaluating the presence or absence of cracks and plastic strain.

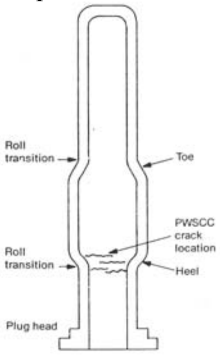

Fig. 2 Schematic of PWSCC cracking in SG plugs

2.2 Strategies for tube extraction

When testing in a high temperature/high pressure water environment, alloy 690 has a general surface film and a thick oxide layer that grows inside the base material. Even though the alloy 690 shows high PWSCC resistance, it has been reported that alloy 690 is more sensitive to SCC than alloy 600 in certain secondary-side environments (e.g. lead contaminated water). [2~4]

Based on the non-destructive inspection record, it is necessary to draw out the tubings suspected of denting, derive the cause and countermeasures for the denting, and derive the correlation with cracks.

In the case of damage caused by wear, the damage rate decreases with time, and the wear damage stops after a certain period of time, but the surface microstructure or stress state is changed due to wear and the thickness becomes thin, so that cracks may occur within or around the damaged area due to wear. The bobbin probe used for non-destructive testing is suitable for detecting volumetric defects such as wear, but it is very difficult to isolate crack defects when wear and crack defects exist together in similar locations.

It was reported that wear damage in the steam generator tubes of the SONGS power plant in the US was caused by thermal hydraulic conditions that were more severe than expected in 2012.[5] In the case of the domestic nuclear power plant to be withdrawn this time, it is different from the case of the US plants, but the same material of alloy 690 was used, and 8 tubes were plugged due to the same degradation factor called wear. It may be necessary to observe whether or not growth into additional defects (SCC) occurs. Therefore, it would be better to use the tubings as a harvesting target in order to check the occurrence of cracks in the wear-damaged area.

The SG channel head divider plate is welded to the channel, and the plates showed some crackings in USA.[6] Fig. 3 shows a schematic of the cracking region at divider plate of SG. [6]

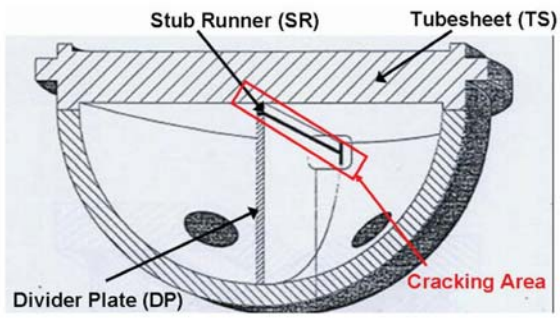

Fig. 3 Schematic of the cracking region at divider plate [5]

Under the above background, the tubing withdrawal method can be summarized as follows.

1) General information: Take out without decontamination and without damaging corrosion products attached to the surface as much as possible.
2) Areas to be withdrawn for analysis of sludge, stress, and temperature effects: 3x3 block (1 place) including tube support plate(TSP) with a lot of sludge, 3x3 block (1 place) including TSP with little sludge, 3x3 block including low temperature tube plate (1 place), SG tubings in the TSP area with a lot of sludge (2 places)
3) Possible denting area: 3x3 block including tube suspected of denting during operation (1 place)
4) Stress corrosion cracking in the tube plugged area: The 3x3 block (2 places) including the mechanical plugged tube and the 3x3 block (2 places) including the welded tube plugging need to be cut out of the entire tube plate
5) Wear defect areas: Two easy-to-draw tubings among unplugged tubings and two easy-to-draw ones among plugged tubings
6) Cracked part of the channel head divider plate welded part: 2 places at the bottom of the tube plate and 2 places for the channel head

2.2 Materials characterization

In terms of corrosion cracking, it can be said that the resistance of alloy 690 has already been verified. [7-11] However, since alloy 690 is a Ni-33% Cr alloy similar to the composition of Ni_2Cr resulting from LRO/SRO, there is much interest in whether alloy 690 will generate such LRO/SRO at the operating temperature of a nuclear power plant and not brittle the material. [12-14]

Although not all researchers agree, some researchers have reported that LRO/SRO can occur when alloy 690 is heat treated for 10,000 to 100,000 hours at a temperature range of 360~450 °C. [15~18]

3. Conclusions

The SG of the power plant subject to withdrawal is made of alloy 690 and has 19 years of operation history until it is shut down in 2017.

In this SG, 15 tubings are plugged by welding and 8 ones mechanical method before installation. In addition, there are 8 tubings that have been plugged due to wear damage exceeding 40% of the tube thickness.

Since this alloy 690 SG tubings experienced high temperature and high pressure operation environment, it can provide important information that can empirically evaluate the corrosion condition of the inner and outer surfaces of the tubings. By accurately analyzing the characteristics of the tube surface oxide and comparing it with laboratory corrosion data, it will be possible to predict the degradation of alloy 690 tubings due to long-term operation.

It would be good if the tubings were drawn out as a standard to check the corrosion characteristics of alloy 690, the denting characteristics of the SG, and cracks in the wear damaged area. It is necessary to check whether cracks occur in structurally weak areas around the SG cannel head divider plate. It is also required to examine whether there is evidence of LRO/SRO in the material operated at around 320°C for 19 years, and a material analysis method is needed for this.

ACKNOWLEDGEMENTS

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