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Causes and Remedies for the Degradation of Steam Generator Tubings in Korean Nuclear Power Plants

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There have been 16 operating nuclear power plants in Korea. A number of degradation mechanisms have been observed to date in steam generator(S/G) tubings, such as primary water stress corrosion cracking(PWSCC), fretting, intergranular attack, secondary side SCC, and pitting. SCC is the cracking-type failure of susceptible material under the combined action of a corrosive environment and sustained tensile stress. PWSCC of Alloy 600 MA tubings was a main reason for Kori-1 S/G replacement, and has become a significant concern for Ulchin-1&2 units. PWSCC has mainly occurred at tubesheet expansion transitions. Kori-3&4 and YGN-1&2 units have experienced fretting and wear of tubes at the anti-vibration bars as a result of tube vibration caused by high cross flows in the U-bend region. Intergranular attack and secondary-side-initiated SCC have occurred at the tube support intersections, in tube sheet crevices, in sludge pies, and recently in the free span between the sludge pile and supports. This attack appears to be caused primarily by caustic. Pitting had occurred in the tubesheet sludge pile regions of retired Kori-1 S/Gs. This form of degradation appears to be an acid attack associated with the introduction of chlorides, sulfur anions, and copper oxides into the steam generator. Much efforts have been made in oder to mitigate the degradation of S/G tubings, such as chemical cleaning, molar ratio control, sleeving, and shot-peening.

Keywords : *Steam generator, tube degradation, PWSCC, ODSCC, shot-peening*

1. Introduction

Steam Generators(S/Gs) in Pressurized Water Reactor (PWR) nuclear power plants(NPPs) are heat exchangers. They transfer heat from a primary coolant system to a secondary coolant system. S/G tubes form a substantial portion of the second fission product barrier in PWRs. For this reason, they must, under all conditions, reliably fulfill this function while retaining sufficient mechanical integrity to preclude the risk of tubes bursting, and thus limit primary-to-secondary leaks to a minimum and ensure a safe shutdown of the reactor. Thus, it is important to establish the structural integrity of the S/G tubings.

S/Gs in PWRs were designed for a 30-40 year operating life. However, the operating experience of these S/Gs shows that the tubings are affected by degradation mechanisms, such as denting, pitting, fretting, intergranular and transgranular stress corrosion cracking(SCC).

Degradation of S/G tubings often requires unscheduled or extended outages for maintenance. These outages are costly in terms of inconvenience, equipment repair, replacement power, and personnel radiation exposure. Continuing problems also increase the probability of steam generator replacement before the end of the power plant design life. The damage being experienced by S/Gs, together with the high costs associated with steam gnenerator repairs, provides a strong incentive to consider S/G replacement. In order to make a decision whether the steam generator replacement is economically feasible or not, first it would necessary to predict the S/G tube degradation in the future.

In Korea, since Kori-1, the Korea's first NPP, started its commercial operation in 1978, KHNP(Korea Hydro & Nuclear Power) have been operating 16 nuclear power plants: 12 PWRs(Kori-1,2,3,&4 units, YGN-1,2,3,&4 units, Ulchin-1,2,3,&4 units), and CANDU type Wolsung-1,2,3,&4 units. The purpose of this paper is to describe the causes of S/G tubing deradation and the remedial measures developed for the Korean nuclear power plants.

2. Degradation modes of S/G tubings

As mentioned above, S/G tubings have prematurely degraded from such mechanisms as denting, pitting, fretting, primary water stress corrosion cracking(PWSCC) and outside diameter stress corrosion cracking(ODSCC).[1)]

These mechanisms are briefly described as follows:

Denting is a circumferential inward movement of S/G tubing at support plate and at tube-sheet locations having open crevices. Distortion of the tubes is caused by the volume expansion of oxides.

Pitting has occurred in the tubesheet sludge pile regions of several PWRs. This form of degradation appears to be an acid attack associated with either the introduction of chlorides, sulfur anions, and copper oxides into the steam generator.

Fretting is caused by vibration of the tube within the tube-support plate clearance. A major concern is that fretted regions are highly sensitive to fatigue cracks. Several units have experienced fretting and wear of tubes at the antivibration bars(AVBs) as a result of tube vibration caused by high cross flows in the U-bend region.

PWSCC is thermally activated, intergranular cracking mechanism. PWSCC has mainly occurred in high stressed regions such as rows 1 and 2 U-bends and tubesheet expansion transitions.

ODSCC, that is, secondary-side-initiated SCC have occurred at the tube support intersections. in tube sheet crevices, in sludge piles, and recently in the free span between the sludge pile and supports. This attack appears to be caused primarily by caustic.

3. Causes and remedies for tube degradation in KNPP

Table 1 presents the S/G model and tube material in korean NPPs. Kori-1's retired S/G tubings were made of Alloy 600MA. It was replaced with new S/Gs with Alloy 690TT tubings. S/G tubings of YGN 3&4, and Ulchin 3 &4 were made of high temperature mill annealed(HTMA) Alloy 600. S/G tubings in Wolsung units were made of Alloy 800TT. Alloy 600TT tubings were installed for the other S/Gs.

As shown in Table 2, various kinds of degradation mechanisms were observed to date, such as denting, pitting, PWSCC, ODSCC and fretting. However, Wolsung-1 did not show any S/G tube degradations to date.

Table 1. S/G model and tubing material in KNPP

Units	Kori-1		Kori-2,3,4	YGN-1,2	Ulchin-1,2	YGN-3,4 Ulchin-3,4	Wolsung-1,2,3,4
	Retired	New					
S/G Model	WH-51	Δ-60	WH-F	WH-F	Fram-51B	ABB/CE	CANDU
Tube Mat'l	Alloy 600 MA	Alloy 690 TT	Alloy 600 TT	Alloy 600 TT	Alloy 600 TT	Alloy 600 MA	Alloy 800 TT

Table 2. Degradation mechanisms for S/G tubings in KNPP

PLANT	Denting	Pitting	PWSCC	ODSCC	Fretting	Remarks
Kori-1 (Retired)	●	●	●	●		- Cu in 2ndary System - Chemical Cleaning in 1990
Kori-2	●			●	●	- Chemical Cleaning in 1993
Kori-3,4 YGN-1,2,3					●	- AVB Vibration
YGN-4				●	●	- AVB Vibration - Circumferential ODSCC
Ulchin-1,2			●			- Shot-peening
Wolsung-1						- No Tube Degradation

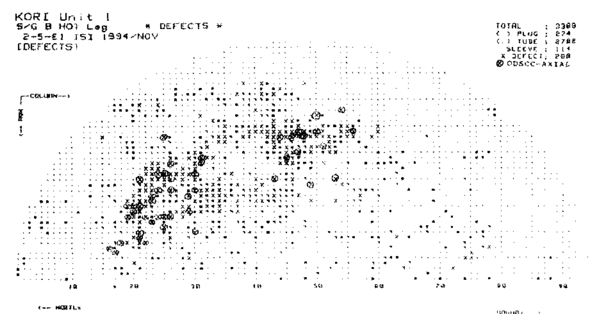


Fig. 1. Defect Tube configuration in retired Kori-1's S/G

3.1 Kori-1 S/G(retired)

Kori-1 stated its commercial operation in April 1978. Kori-1 is a two-loop plant having a Westinghouse NSSS with 51 Series steam generators with mill anneald Alloy 600 tubings. This plant has a nominal design output rating of 595 MWe. S/G tubings

were severely damaged as shown in Figure 1. S/G replacement decision was made by technical and economical evaluations.

Figure 2 shows the results of S/G tube degradation

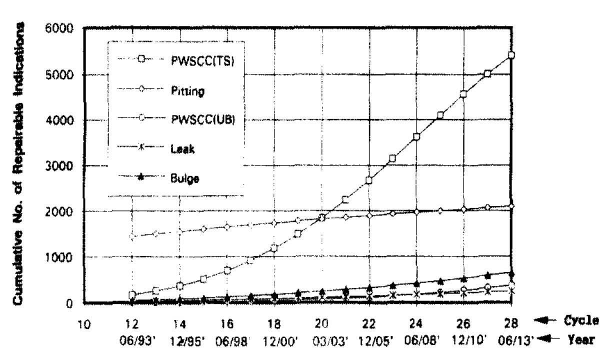


Fig. 2. Degradtion predictions for retired Kori-1's S/G tubings[2)]

predictions.[2] This Figure shows the projections for each degradation modes. Pitting degradation, which has been the most severe degradation mode in Kori-1, would be slowed down, and could be modeled at a reduced rate, due to the beneficial effects of the chemical cleaning. On the other hand, PWSCC(TS) would rapidly be increased and would be the major tube degradation mechanism in the future. The other degradations would be continued, at a relatively slow rate.

The economic evaluation results[3] indicates that the SG replacement as early as possible is the most profitable strategy, especially with power uprating say about 5%. In the assumed plant life extension beyond design life of 30 years, immediate replacement appears to be more cost effective alternative. Therfore decision was made to replace the Kori-1 S/Gs in 1998, taking into account the economic evaluation results, together with the operation reliability and public acceptance.[3]

It was revealed that pitting was occurred due to condenser leakage, high dissolved oxygen and copper oxides sludge. In order to prevent pitting, copper condenser tube was replaced with titanium in 1988. Hydrazine(H_2NNH_2) was used for water chemistry control to reduce dissolved oxygen.

PWSCC in Kori-1 was resulted from the Alloy 600 MA susceptibility, high temperature induced by superheating at sludge pile, and high residual stress at roll transitions. Since 1990, KHNP have performed chemical cleaning to remove the sludge, which could possibly mitigate the superheating effect.

There was no majpr tubing corrosion on the secondary side of S/G until November 1994, when a large number of ODSCC developed and caused an unscheduled outage. Remedial measures were developed with the goal of avoiding further unscheduled outages caused by ODSCC.[4] The remedial measures included: (1) crevice neutralization prior to starting by crevice flushing with H_3BO_3 and molar ratio control during operation using NH_4Cl, (2) corrosion potential reduction by hydrazinesoaking and suppression of makeup water oxygen below 20 ppb to avoid copper oxide formation, (3) TiO_2 inhibitor applied by preoperation soaking, and (4) a hot-leg temperature reduction of 5℃. With the remedy program, no leakage occurred during the 3.5 years prior to the S/G replacement in 1998.

3.2 Kori-2 S/G

Kori-2 has been in commercial operation since July 1983. In Kori-2, fretting damage due to vibration between AVB and S/G tubings have been observed. AVB wear is not typically a life-limiting S/G tubing degradation mechanism, and is typically handled by occasional tube plugging. The previous study,[5] in which the rates of progression of AVB wear at the Westinghouse Model F-type units were determined using the industry experience of the S/Gs, showed that the Model F S/Gs would reach 4.5% tubings repaired due to AVB wear at a hypothetical end of life of 34.7 EFPYs. No remedial measures have been used to limit the AVB wear degradation.

ODSCC was detected, which was caused by the stress augmentation due to denting and high hot leg temperature. Pb-induced TGSCC was also observed. In order to prevent the ingress of impurities(e.g. Pb), condensate polisher has been operated in tight control.

3.3 Kori-3&4 and YGN-1,2,3&4

Only fretting wear was observed in these plants. No remedial measures have been used to limit the AVB wear degradation. In YGN-4's S/G tubings, circumferential ODSCC was detected from ECT examination in 2001. Temporary conclusion is that ODSCC was resluted from stresses at roll overlaps and denting. The damaged tubings were sleeved with stabilizer. Boric acid soaking and molar ratio control have also been planned. In long-term plan, sludge lancing and ORT(Opeartion at Reduced Temperatue) have been considered.

3.4 Ulchin-1&2

Ulchin-1&2 have been in commercial operation since Sept. 1988, and Sept. 1989 respectively. The three recirculating S/Gs with FRAMATOME 51-B series were constructed with thermally treated Alloy 600 tubes. These tubes were expanded over the full depth of the tube sheet by mechanical rolling with a kiss rolled transition zone. Specifically targeted motorized rotating pancake coil inspections in 1994 confirmed the presence of a significant number of tubes with axially oriented PWSCC in roll transition zones. In February 1994, shot peening had been employed in order to mitigate the PWSCC damage after five cycles of operation.

4. PWSCC and shot-peening

The following analyses have been performed to clarify the effects of compressive residual stress induced by shot peening on PWSCC, based on the Ulchin-1 nuclear power plant data.

The previous study showed that the shape of the cracks is different for standard and kiss roll transitions.[6] Standard roll transitions without kiss roll exhibited half elliptical, semi-elliptical, very regular for short as well as long cracks. On the other hand, roll transitions with kiss-roll presented elliptical shape for short cracks, and complex "car shape" for long cracks.

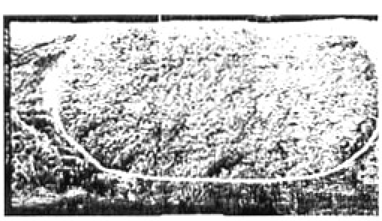
Figure 3 presents the cracks observed before shot-peening.[7] Since Ulchin S/G tubings were kiss-rolled, elliptical crack shape was expected for short cracks. However short crack exhibited semi-elliptical shape, which is expected in standard roll transition. On the other hand, long cracks exhibited car shape, as expected in kiss-rolled transitions. This may indicate that kiss-rolling is not effective on the crack propagation when the crack is short. As cracks grow longer, the crack propagation becomes affected by kiss-rolling.

Figure 4 shows the crack configuration after shot-peening. Short cracks exhibited elliptical shape. The shape of long cracks are also elliptical, but simply elongated in the axial direction. These observations indicate that the stress state of the crack front has been changed after the application of shot-peening.

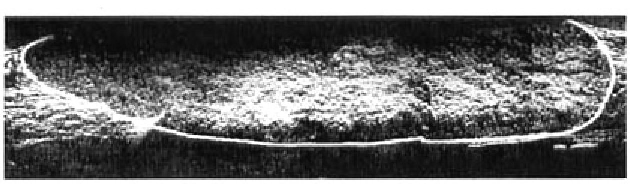
(a) R4C33(Crack length: 2.39 mm, Max. Depth: 0.58 mm)

(b) R4C33(Crack length: 5.88 mm, Max. Depth: 1.18 mm)

Fig. 3. Crack shape before shot-peening



(a) R4C39(Crack length: 2.1 mm, Max. depth: 1.07 mm)



(b) R4C30(Crack length: 6.15 mm, Max. depth: 1.09 mm)

Fig. 4. Crack shape after shot-peening

Table 3. Average crack length for each cycle in Ulchin-1 S/Gs

Cycle	5^{th}	6^{th}	7^{th}	8^{th}
Crack Length	3.88 mm	3.44 mm	3.36 mm	3.25 mm

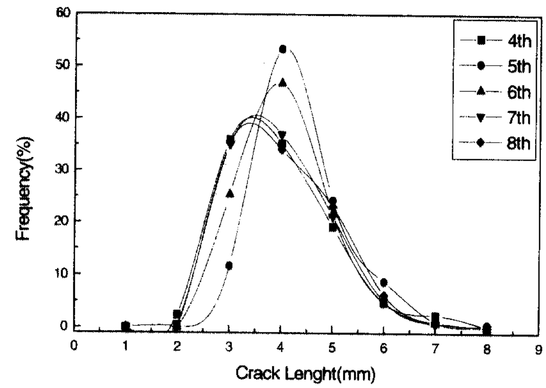


Fig. 5. Time history of the distribution of crack length in Ulchin-1[9]

Table 3 shows the average crack length before and after shot peening. Shot-peening was performed in 5th cycle. Average crack length appears to keep decreasing after shot-peening. Average crack length of the 8th cycle is 3.25 mm, which shows 16% decrease compared to 3.88 mm of the 5th cycle before shot-peening.

Figure 5 illustrates a typical time evolution of crack length distribution, which predict the deformation and shift of the initial distribution, over several successive operation cycles, as predicted by Hernalsteen's model[8] The crack length distribution histogram has an overall bell shape. With time, the length distribution curve shifts towards the range of longer lengths.

However in Ulchin-1, the distribution curve skewed to the shorter crack lengths. This implies that the crack growth is retarded after shot-peening. Therefore, the number of short cracks were increased while the growth rate of long cracks were slowed down.

These results show that it is very clear that crack growth in the axial direction was retarded by residual compressive stress layer. The crack growth retardation effect by shot-peening is obvious.

5. Conclusions

In Korean nuclear power plants, various kinds of degradation mechanisms have been observed to date in steam generator tubings, such as denting, pitting, PWSCC, ODSCC, and fretting.

In recent years, the cracking-type failures, such as PWSCC and ODSCC, have become a significant concern for steam generator tubings in KNPPs. The remedial measures for ODSCC mitigation, such as hydrazine soaking, TiO_2 application, and a hot-leg temperature reduction was believed to be effective.

Shot-peening did not appear to be very effective when it was applied in the middle of commercial operation. The residual compressive stress induced by shot-peening resulted in crack growth retardation in the axial direction.

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