ABSTRACT

The degradation of steam generator tubes made of alloy-600 continues to be a serious problem since the first nuclear power plant in Korea began commercial operation in 1978. The steam generator of the oldest nuclear power plant was replaced due to various defects and tremendous expenses for maintenance after its 20 year operation. In addition to the alloy-600 tubes in steam generator, alloy-600 butt welds in major components or J-weld in tube penetration have been a concern for the regulatory body and utilities due to their frequent failures not only in Korea, but around the world.

The Korean regulatory body has put forth continuous efforts to manage alloy-600 degradation since the steam generator tube rupture event in 2002. An enhanced inspection program was introduced to monitor the integrity of the whole steam generator closely. Performance demonstration system for ultrasonic test (UT) and eddy current test (ECT) has also been initiated from 2002. Still, some alloy-600 drain line tubes or tube plug failures have occurred in nuclear power plant in Korea.

This paper aims to introduce the regulatory bodies’ efforts to manage the alloy-600 degradation such as round robin test of retired steam generator tubes, the destructive test of parts and performance demonstration (PD) program for non-destructive examination (NDE) in Korea.

Keywords: PWSCC, Alloy-600, Round Robin Test, Materials Degradation, Performance Demonstration, Dissimilar Metal Weld

INTRODUCTION

Since the first nuclear power plant’s commercial operation in 1978, the degradation of steam generator tubes and the primary water stress corrosion cracking (PWSCC) problems of alloy-600 welds have continued a great concern and the regulatory body and the utility have made a lot of efforts to manage the degradation of alloy-600 components.

There are various types of a steam generator at the nuclear power plants in Korea. Some degradation mechanisms tend to be found in specific steam generator models, different tubing materials and position in the steam generator. Emphasis is placed on the design aspects and fabrication methods which may affect steam generator degradation. In the WH model F steam generator tubes made of alloy-600TT, corrosion-related degradation such as denting, pitting, stress corrosion crack (SCC) and wear problems has been experienced. In the CE system 80 steam generator tubes made of alloy-600HTMA, SCC has been detected under sludge pile on the top of tubesheet (TTS) and some wear defects also detected around structure like batwing and vertical strap.

During the period of year 1999-2006, many PWSCCs were found in alloy 82/132/182 butt welds of PWR plants around the world. On October 7, 2000, a large quantity of boron was identified on the floor and protruding from the air boot around the “A” loop RCS hot leg pipe in VC Summer Nuclear Power Plant in USA. Accordingly, UT, ECT and VT (visual test) were applied, and an axial crack-like indication was identified. The hot leg weld was cut out and destructively tested. The indication was determined to be an axial crack approximately 2.5 inches long and almost through wall crack which was caused by PWSCC [1].

At Wolf Creek in October 2006, three indications were found in the pressurizer surge nozzle-to-safe end weld, and two separate indications were in the safety and relief nozzle-to-safe end welds. These findings paid significant attention to the current inspection schedules and plans. According to the USNRC requirement, the baseline inspection of pressurizer for the same type of nuclear power plant was to be finished by spring 2008 [2].

To assure the steam generator tube integrity, nuclear industries in Korea have performed the round robin test of ECT of a retired steam generator. Also, the Korean regulatory bodies have required
carrying out the performance demonstration of UT for DMW of pressurizer to assure the integrity of DMW as well as for the welds of S/S and C/S piping.

ROUND ROBIN TEST FOR ALLOY-600 STEAM GENERATOR TUBES

**ECT mockup test**

The eddy-current methods have some limitations in detecting degradation damage to steam generator tubes. The depth sizing capabilities for these inspection methods are particularly limited. In some cases, the uncertainties in depth sizing of the defects are being determined by comparing the eddy-current measurements with the actual known size.

To demonstrate the round-robin test, a mock-up test was conducted by commercial in-service inspection (ISI) teams prior to round-robin tests on degraded steam generator tubes removed from service. There were 30 tubes made of Alloy-600 MA in the mock-up, which contain 34 circumferential EDM (Electrical Discharge Machining) notches and 51 axial EDM notches in tubes. In this study, both of signal amplitude based depth sizing method and phase angle based depth sizing method using motorized rotating pancake coil (MRPC) probe were used to compare the measured size of notches in the tubes with actual ones. In general, the nuclear industry indicates that ECT data for pulled out tubes show less scattered results for phase angle based sizing method than for amplitude based sizing method. However, there were some limitations, in particular, when several closely spaced axial notches existed. Furthermore, phase angle based sizing method did not work well in the case of circumferential ODSCC at hard-rolled intersections. Based on this study, it could be concluded that the amplitude-based depth sizing is more reliable than the phased-based depth sizing for plus-point ECT probe if the cracks are similar to the EDM notches [3].

**ECT round robin test using retired S/G tubes**

After tube rupture incident of Uljin unit 4 in 2002, a national research project for the round robin test (RRT) of S/G tubes was initiated to improve the reliability of ECT using actual degraded tubes of S/G retired in 1998 from the oldest Korean nuclear power plant. This S/G had 347 plugged and 488 sleeved tubes among 3388 tubes. The utility, ISI teams and a research institute participated in this project. The tubes for inspection were selected after reviewing the inspection and maintenance data during operation of the steam generator. All the plugs were removed from the tubes to be examined because the tubes for RRT had been plugged for stopping the tubes’ use during the operation period. The inspections were performed using bobbin coil, MRPC probe, X-probe and intelligent probe for the top of tube sheet (TTS), U-bend region, and first tube support plate (TSP) region. After the ECT for selected tubes, the destructive examination was carried out in Semi-hot lab for verification of the ECT results. Table 1 and 2 show the numbers of selected tubes for ECT and the numbers of tubes for destructive examination (DE).

<table>
<thead>
<tr>
<th>Technique</th>
<th>Organization</th>
<th>Inspected tubes</th>
<th>Sum</th>
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<tbody>
<tr>
<td>Bobbin</td>
<td>KPS</td>
<td>598</td>
<td>598</td>
</tr>
<tr>
<td>RPC</td>
<td>TTS</td>
<td>796</td>
<td>1590</td>
</tr>
<tr>
<td></td>
<td>KPS</td>
<td>252</td>
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<td></td>
<td>1st TSP</td>
<td>KPS</td>
<td>542</td>
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<tr>
<td>X-probe</td>
<td>R/D Tech, NEL</td>
<td>759</td>
<td>759</td>
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<tr>
<td>I-probe</td>
<td>Mitsubishi, Zetec</td>
<td>532</td>
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Table 1 - Numbers of selected tubes for ECT
For verification of the detection ability, the probability of detection of defects was conducted. POD was analyzed with log logistic function. For axial crack in the tube inside (ID) and outside (OD), the POD function was derived from the variables such as maximum depth, effective depth, effective length and fractured pressure. From the logistic POD function analysis, the crack depth is major variable in POD and the crack length affects the POD of defects. Fig. 1 shows the POD of effective crack depth for 9 evaluation teams and Fig. 2 shows the average POD for effective and maximum crack depth and the POD for maximum crack depth with MRPC, I-probe and X-probe [4].
To check the reliability of ECT crack sizing, all the different kinds of cracks such as OD and ID axial cracks as well as OD and ID circumferential ones in different regions of the retired tubes were examined. Fig. 3 shows an example of the RRT results of OD axial crack sizing. Most of cases showed that ECT results underestimated the actual effective crack depth similar to Fig. 4. Fig. 5 shows that actual OD axial cracks at TTS are much deeper than ones at TSP. In these regions, multiple axial cracks were detected and could not be characterized precisely by ECT [5].

The RRT results will be utilized in the development of optimized procedure for ECT after peer review and applied to upgrade the steam generator management program (SGMP). This upgraded SGMP will be reviewed by regulatory body in the near future.

**Figure 3 - Comparison between ECT results and DE results for OD axial cracks.**

**Figure 4 - Details of a circumferential crack**
Figure 5 - Depth vs. length of OD axial cracks at TTS and at TSP

ALLOY-600 MATERIALS ISSUES IN KOREA

PWSCC on S/G drain nozzle

Boric acid deposits were detected on drain nozzle of S/G cold leg side of OPR-1000 in 2007. Several PT indications were revealed on the inside of this nozzle like in Fig. 6. Even though drain nozzles of S/G hot leg side were replaced with alloy-690 prior to start-up of this plant reflecting the head penetration crack experience at Bogey plant in France, drain nozzles of S/G cold leg side were not replaced because PWSCC was not expected to occur in low temperature at that time. After this incident, all the S/G nozzles made of alloy-600 were requested to be examined or replaced with PWSCC resistant materials.

Figure 6 - PT indications on S/G drain nozzle and configuration of S/G channel head

Several indications on an instrument nozzle of S/G were detected by ECT, and verified by UT. To check the reliability of non-destructive examination (NDE), DE was required after cutting out of the nozzle. Fig. 7 shows the NDE and DE results for an instrument Nozzle of S/G. All the remaining nozzles were required to be replaced with PWSCC resistant materials as soon as possible because minor cracks could not be detected by current NDE [6].
Figure 7 - NDE (UT/ECT) and DE Results for an instrument Nozzle of S/G

Maintenance and Performance Demonstration for Dissimilar Metal Weld

In Korea, the starting day of PD for UT and ECT was defined in MEST (Ministry of Education and Science Technology) Notice 2004-13. PD for the carbon and stainless steel piping and bolts/studs was started from 2005. PD of DMW (dissimilar metal weld) was supposed to start in June 2006. However, the utility requested for the preparation time to establish DMW PD system and specimen. Therefore, DMW PD system was prepared in two steps by the utility. In the first step (from June 2007 to Nov 2010), the utility used DMW PD certificate holder, equipments and procedures that were approved by US EPRI. During this step, Korean PD center will prepare DMW mock-ups including plant specific mock-ups and develop the examination technique for DMW by R&D and international cooperation such as PINC (program for the inspection of nickel components) and EPRI technical guidance. Also, KINS required KEPRI which is a Korean PD center to establish DMW PD system including plant specific mock-ups until Nov. 2010. Currently, all nuclear power plants including PRZ DMW were inspected according to alloy-600 management program of the utility and no cracks have been found up to now. In the second step (after Nov. 2010), the utility will use Korea PD system. Fig. 8 shows the PD organizational structure in Korea.
The schematic design of weld overlay is shown in fig. 7. For mitigation, pressurizer DMW of Kori unit 1 will perform full structural weld overlay (FSWO) during 2009-2010. This overlay was designed to improve the inspectability and the examination coverage. All nuclear power plants having Pressurizer DMW will perform FSWO step by step until 2014. KINS will review and verify the structural analysis and the repair process in detail because the utility already submitted a topical report related to FSWO to the regulatory body. Vendors are developing the weld overlay technique and UT technique as shown in Fig 10 and 11 respectively.

Figure 9 - Full structural weld overlay design of pressurizer surge nozzle

Figure 10 - Photos of R&D for weld overlay.

Figure 11 - Development of UT technique for weld overlay in a company (KPS).
SUMMARY

The Korean regulatory body has put forth continuous effort to manage alloy-600 degradation such as an enhanced inspection program, the regulation for ECT qualified data analysis, the adaptation of steam generator management program and the introduction of PD system for DMW. Preventive maintenance could be done to a lot of defective tubes through the enhanced inspection program for the S/G tubes and the reliability of ECT technologies would be more improved reflecting the results of the RRT used a retired steam generator. Integrity of S/G would be improved by the upgrade of steam generator management program. FSWO will be conducted to mitigate the alloy-600 DMW degradation of pressurizer nozzles and the PD system for DMW and structural weld overlay will be established in the near future. These regulatory efforts will improve the safety and the reliability of pressure retaining components.

REFERENCES

1) NRC Information Notice 2000-17 : Crack in Weld area of Reactor Coolant System Hot Leg Piping at V. C. Summer, United States Nuclear Regulatory Commission, (2000)
5) Seong Sik Hwang, Jangyul Park, Hong Pyo Kim, Database development on SCC from a retired steam generator, Transactions of the Korean Nuclear Society Spring Meeting, PyeongChang, Korea, (2008)