OPERATING EXPERIENCE WITH THERMALLY TREATED ALLOY 600 TUBES IN MODEL F STEAM GENERATORS: CRACKING

M. Kim, C. Yim, and J. Park
Kaitec, Daejeon, Korea

Abstract: Steam generator tube in nuclear power plant is a boundary between primary side and secondary side, whose integrity is one of the most critical factors to nuclear safety. Mill annealed Alloy 600(Alloy 600 MA) tubes which had been used for the first generation steam generators prior 1980 were weak in stress corrosion. To overcome this weakness, thermally treated Alloy 600(Alloy 600 TT) tubes have been introduced to the second generation steam generators after 1980. Alloy 600 TT tubes have performed well in service to date without causing serious problems except some special incidents in the U.S[1, 2]. But, Alloy 600 TT tube crackings have emerged at the hotleg top of tubesheet in some Westinghous F model steam generators in Korea recently, which needs study for the reason and continuous attention. This paper introduces those tube crackings with related data, observes each cracking's characteristics, and looks into their environmental differences, concluding possible causes for them.

Introduction: It was April in 1978 when the first nuclear power plant in Korea, Kori 1, went into commercial operation. Kori 1 had 2 Westinghouse 51 steam generators with mill annealed alloy 600 tubes and experienced serious corrosion. Because of this corrosion problem in alloy 600 mill annealed tubes, later 5 plants in Korea employed thermally treated alloy 600 tubes which are known to have more resistance to corrosion to their steam generators. Steam generator design had also been changed from W-51 to model F. Kori 2, Kori 3, Kori 4, Younggwang 1 and Younggwang 2 are these plants and have performed their service well until now as other steam generators with Alloy 600 TT in the U.S. have except Seabrook[1] and Braidwood[2], both of which experienced ODSCC at tube supports from excessive tube residual stress during stress relief process. Major tube degradation mechanism in these Westinghouse model F steam generators in Korea had been fretting wear at anti-vibration bar location. But recent eddy current inspection results reveal that the circumferential ODSCC(outside diameter stress corrosion crack)'s at the hotleg top of tubesheet have occurred. Before looking into these crack indications, here we present each plant's information in brief.

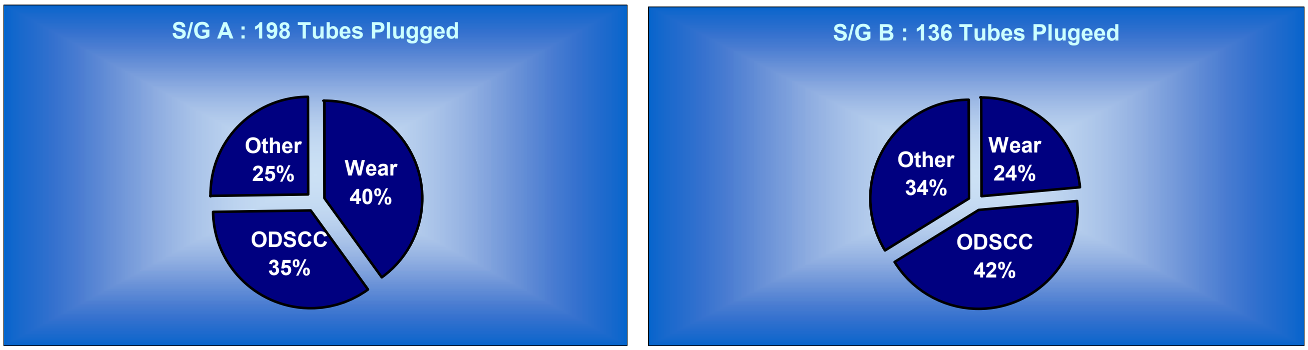
Operational History. There are 5 plants with Westinghouse model F steam generators – Kori 2, Kori 3, Kori 4, Younggwang 1 and Younggwang 2. Kori 2 has 2 loops of steam generators and the others have 3 loops. All are PWR type with capacity of 950 MW except Kori 2(650MW) and manufactured by Westinghouse. Because many nuclear power plants with A600MA tubes including prior unit Kori 1 had experienced lots of corrosion problems e.g. cracks and pittings, tube material has changed to A600TT. Tube expansion method has also been changed with hydraulic full-depth expansion, which is known to have less residual stress than mechanical rolling or explosive expansion. Table 1 shows the summary of these steam generator information.

Table 1. Status of the Model F Steam Generators in Korea

Plants	# of SG's	Comm. Ops	Reactor Type	Capacity (MW)	Model	Manufacturer	Tube Material	# of Tubes /SG	Expansion Method
Kori 2	2	07/83	PWR	650	F	Westinghouse	A600TT	5626	Hydraulic
Kori 3	3	09/85	PWR	950	F	Westinghouse	A600TT	5626	Hydraulic
Kori 4	3	04/86	PWR	950	F	Westinghouse	A600TT	5626	Hydraulic
Younggwang 1	3	08/86	PWR	950	F	Westinghouse	A600TT	5626	Hydraulic
Younggwang 2	3	06/87	PWR	950	F	Westinghouse	A600TT	5626	Hydraulic

Results: Kori 2. Kori unit 2 started its commercial operation in July 1983 and has completed its 18th ISI in October 2003. It had been expected to show better performance against to corrosion, but 105 tubes had to be repaired because of dent-assisted stress corrosion cracking after the operation of just 4 cycle. The secondary side chemistry control was poor until the 5th cycle, and the corrosion of iron oxide in the sludge caused dents, resulting in both circumferential and axial cracks originated from ID and OD at the top of tubesheet. In 1990, pulled tube examination was performed and root cause for the SCC's was evaluated[1]. Later in 1992, chemical cleaning according to EPRI *Fe* removal process was performed. It was the exceptional case that occurred SCC's in thermally treated alloy 600 tubes with hydraulic expansion and showed how fast caustic environment could accelerate stress corrosion crack in spite of improvement of tube material and steam generator design. The SCC's after chemical cleaning are mostly the effect of cumulated operation years. Fig. 1 shows tube repair status after 18th ISI[2].

Figure1. Tube Repair Status of Kori2(Oct. 2003)



Kori 3 & 4. Both units completed their 14th ISI – Kori 3 in December 2002 and Kori 4 in April 2003. As it's known, fretting wear is major degradation mechanism in both units. In case of Kori 3, there were 3 records of stress corrosion crack occurrence. The first one was OD circumferential crack at the hotleg top of tubesheet found in the tube R14 C61 of steam generator 'C' in 1995. According to the final report for 1995 inspection, this indication had characteristic of 270 degree width in circumferential direction and about 40 volts in amplitude. It seems to be clear that this one looked like a big circumferential crack, but it's hard to believe that this kind of severe crack had existed without any leakage considering 100% drilled hole on the calibration standard is set up as 20 volts. The second one was also OD circumferential crack found in the tube R52 C88 of steam generator 'B', but the location was at the coldleg top of tubesheet in 1999. This was a miscall of volumetric indication caused by loose part. The last and the most recent one was ID axial crack at the hotleg top of tubesheet found in 2002. All three tubes data were supposed to be peer-reviewed, but the data for R14 C61 was not available at that time. So, the latter two tubes were peer-reviewed and concluded R52 C88 to be not a circumferential crack but a volumetric indication, and R14 C52 to be NDD(no detectable degradation). Therefore, it's become that there is no SCC exists in Kori 3 and 4 yet.
Younggwang 1 & 2. Younggwang 1 has completed its 14th ISI in May 2003 and Younggwang 2 has completed its 13th in January 2003. Major degradation in both units is wear. Younggwang 1 has no SCC yet, but 4 SCC's have been found at the hotleg top of tubesheet in Younggwang 2 in 2003. The tube ID's are R4 C98 from S/G 'B', R41 C33, R39 C43 and R48 C44 from S/G 'C'. These tubes have not been tested with rotating probe until then. Inspection scope for the RPC test of hotleg tubesheet expansion used to be just 300 tubes covering sludge zone, but it's expanded to 100% of TTS hotleg for all model F and KSNP steam generators since August 2002 after tube rupture accident of Ulchin 4 in 2002. Ironically, thanks to the inspection scope expansion, these cracks could be found, but historical compare with RPC data is not available. And we cannot be sure from when these cracks have existed. There was no leakage before these cracks were detected, and all the 4 tubes were repaired by plugging[3]. No further alternative test like ultrasonic test or visual test to confirm these crack indications was conducted.
These 4 tube data were peer reviewed with Kori data by EPRI in late June last year, and results confirmed that all are OD circumferential cracks except R39 C43. There was a discrepancy whether this indication meets flaw criteria or not and concluded as NQI(non-quantifiable indication). Table 2 indicates peer review results for Kori 3&4 and Younggwang 1&2.

Table 2. Peer Review Results

Plant	S/G	Tube ID	Damage	Loc	Yr	Peer Review Results	Comments

Kori 3	B	R52 C88	ODSCI	TSC	1999	Meets flaw criteria, but SVI	Unanimous
Kori 3	C	R14 C52	IDSAI	TSH	2002	Does not meet flaw criteria	Unanimous
YG 2	B	R 4 C98	ODSCI	TSH	2003	Meets flaw criteria	Unanimous
YG 2	C	R 41 C33	ODSCI	TSH	2003	Meets flaw criteria	Unanimous
YG 2	C	R48 C44	ODSCI	TSH	2003	Meets flaw criteria	Unanimous
YG 2	C	R39 C43	ODSCI	TSH	2003	Non-Quantifiable Indication	Not unanimous

Eddy Current Data Review from Younggwang 2. The RPC test data for the 4 tubes from Younggwang 2 are presented in Fig. 2 to Fig. 5, and Fig. 6 provides the flawed tubes location on tubesheet map. Note that *the flawed tubes are not in the conventional sludge zone*. Bobbin data of the recent inspection for the 3 tubes (R4 C98, R41 C33 and R48 C44) show low sludge height of about 0.5 inches, but R39 C43 seems to have partial deposit from the top of tubesheet up to 2.5 inches. The prior inspection(12^{th}) sludge measurements results indicate quite high value of 3 ~ 4 inches for all 4 tubes[4], but it seems to have been soft sludge easily removed by secondary side lancing judging from low sludge height in recent inspection results. And these SCC's differ from those found in Kori 2 in that they have no dents at the expansion transition.

Figure 2. RPC Data: SCI of R4 C98(S/G B)

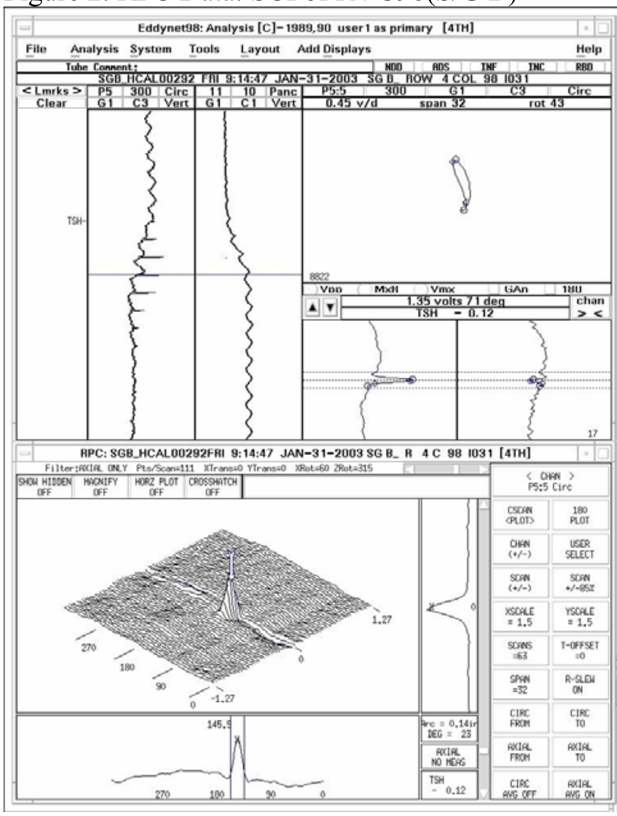


Figure 3. RPC Data: SCI of R41 C33(S/G C)

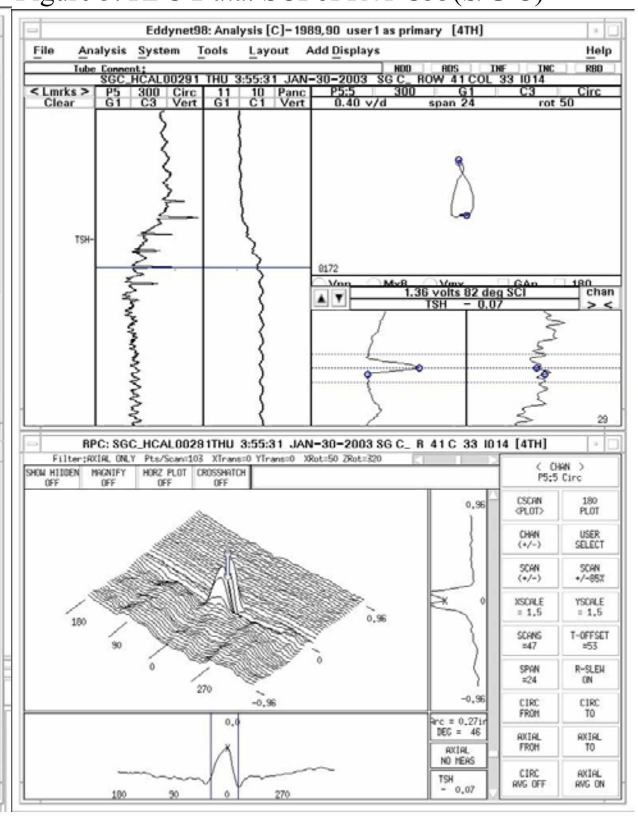


Figure 4. RPC Data: SCI of R48 C44(S/G C)

Figure 5. RPC Data: NQI of R39 C43(S/G C)

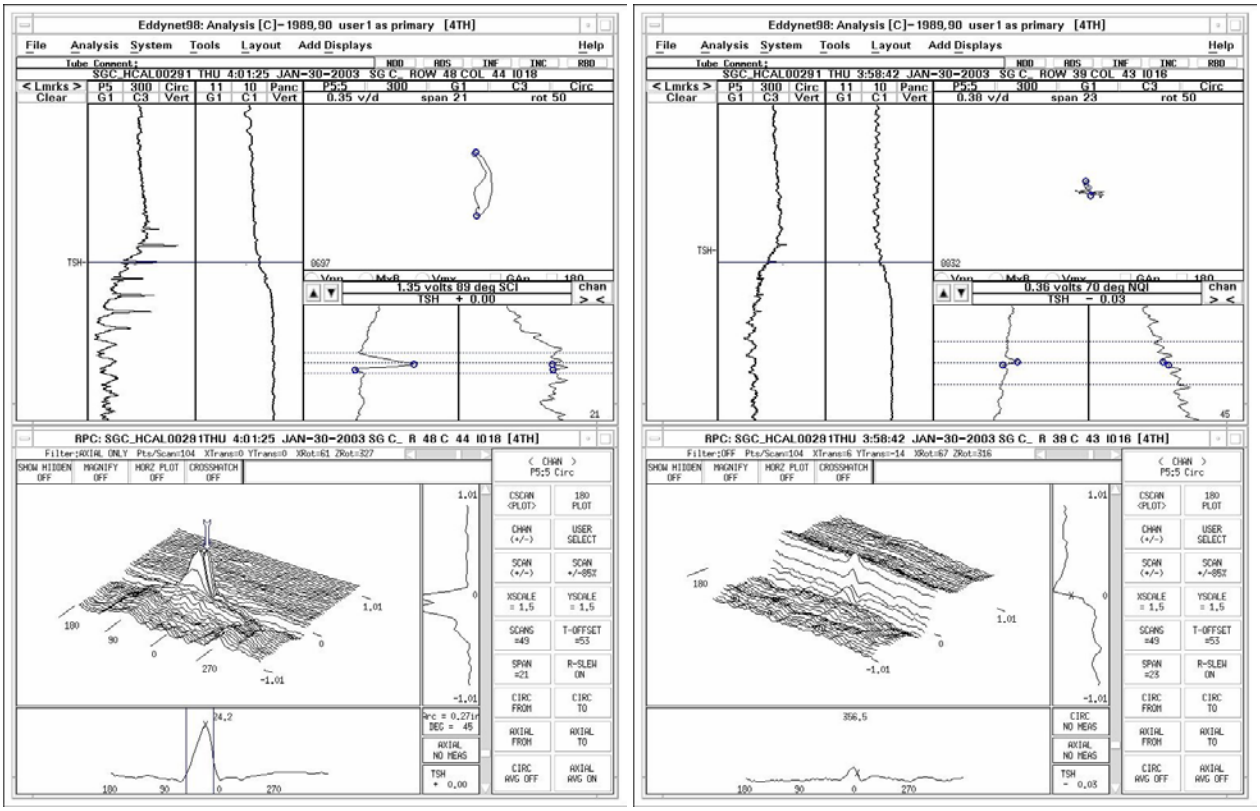
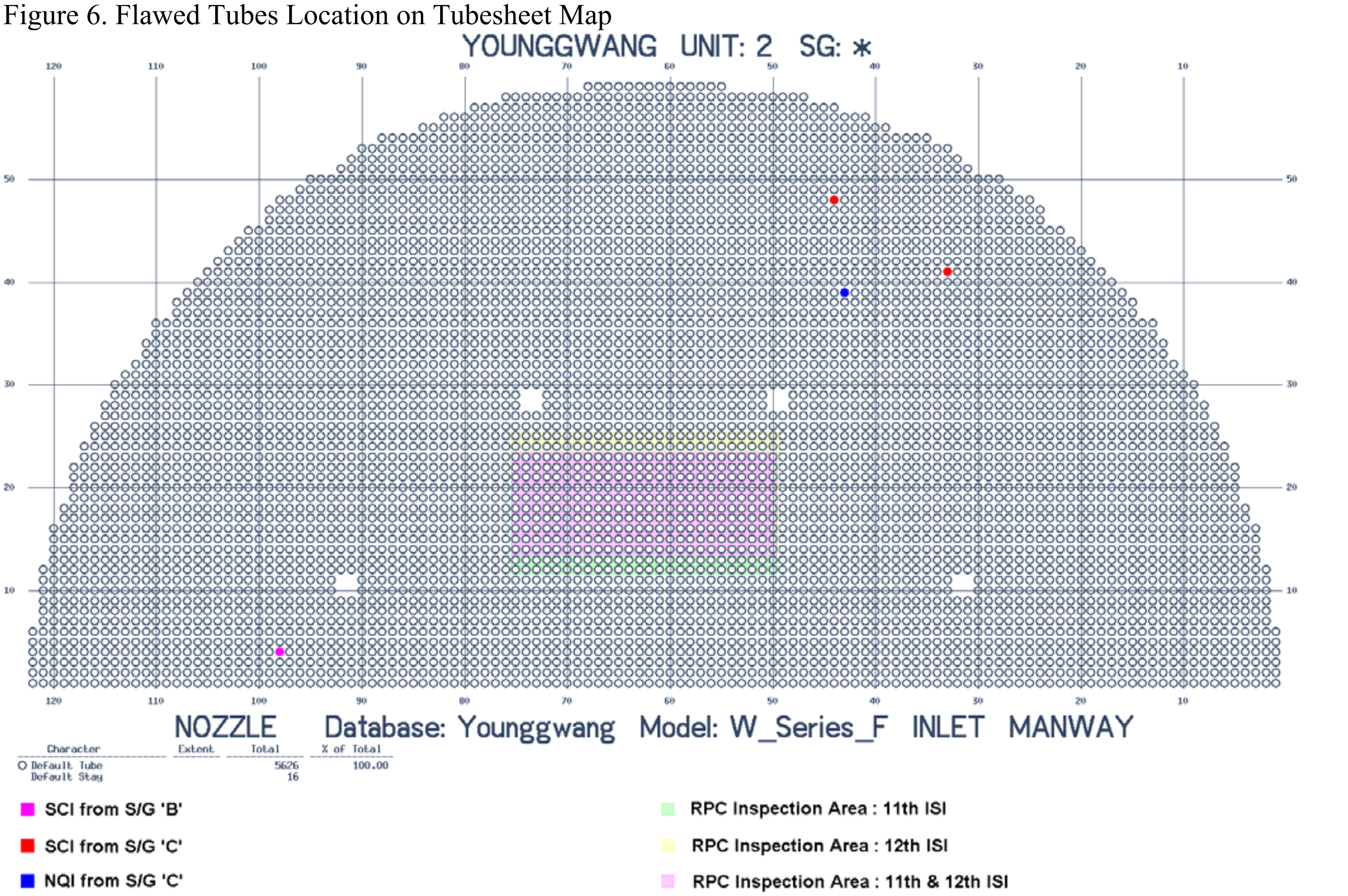


Figure 6. Flawed Tubes Location on Tubesheet Map



Discussions: EPRI Degradation Prediction Model. EPRI has issued report on prediction method of A600TT tube performance from A600MA field data results addressing A600TT tube material and steam generator design changes in September 1997[5]. Papers on this model have been published and presented at various meetings, and several degradation assessments using the EPRI model are being reported.

The EPRI prediction method is based on field data from A600MA and adjusted to the laboratory data results. It uses a Weibull statistical distribution for A600MA degradation and improvement factor for materials and other design features. First, A600MA field corrosion data from several plants are normalized to a standard operation temperature T_{hot}(618°F), and start time and progression of degradation are determined. Then, improvement factors for tube material change, TSP material change, steam generator design change are developed respectively(See Table 3) and applied to establish a corrected start and progression of degradation for A600TT tubes.

Table 3. EPRI Selected Improvement Factors

<table>
<tr><th>Improvement Factor(IF)</th><th>A600TT</th><th>A690TT</th></tr>
<tr><td>ODSCC Materials IF</td><td>2.0</td><td>4.0</td></tr>
<tr><td>PWSCC Materials IF</td><td>2.1</td><td>immune</td></tr>
<tr><td>PWSCC Hydraulic IF</td><td colspan="2">1.2</td></tr>
<tr><td>ODSCC Hydraulic IF</td><td colspan="2">1.0</td></tr>
<tr><td>Stainless TSP Materials IF</td><td colspan="2">1.0</td></tr>
<tr><td>Quatrefoil TSP Design IF</td><td colspan="2">1.5</td></tr>
</table>

For TTS circumferential ODSCC, EPRI Weibull Median time to reach 0.03% of feedring steam generator tubes repaired is 6.12 EFPY. This median time is multiplied by EPRI estimated improvement factor 2.0 and yields 12.24 EFPY(See Fig. 7). In case of Model F steam generators, each steam generator has 5,626 tubes, so repaired fraction of 0.03% becomes 2 tubes per steam generator. This seems to give a quite good explanation for Younggwang 2(See Table 4). The EFPY values in the table are calculated assuming T_{hot} is 618°F the standard operation temperature, which is the actual operation temperature or a little bit lower.

Figure 7. EPRI Weibull Projection

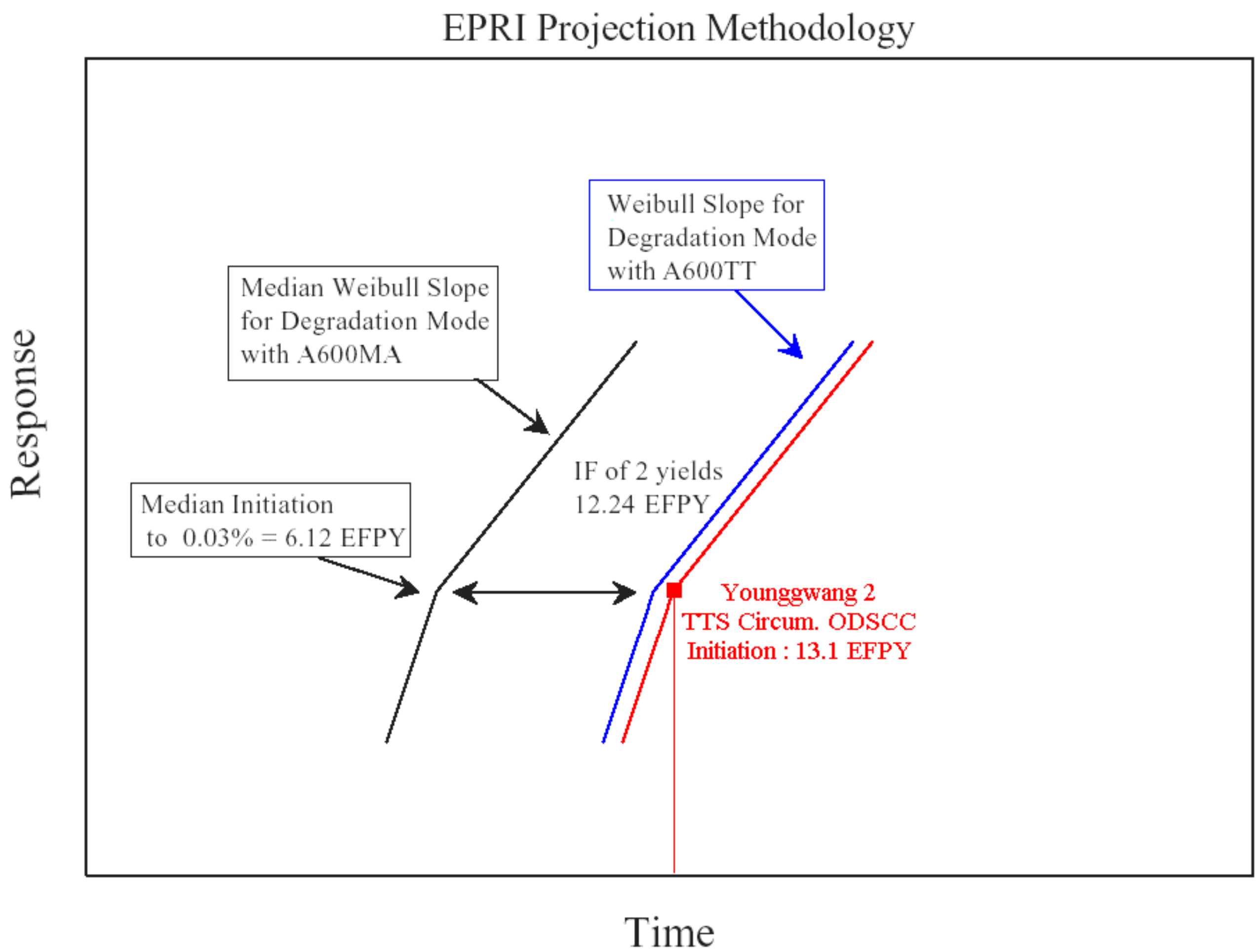


Table 4. Estimated EFPY for Model F S/G's in Korea

Plants / EFPY	Kori 2	Kori 3	Kori 4	Younggwang 1	Younggwang 2
Recent ISI (month/year)	17th(10/02)	14th(12/02)	14th(04/03)	14th(05/03)	13th(01/03)
EFPY	16.2	13.9	13.2	14.4	**13.1**

But, if these cracks are the result of tube aging, why did they occur only in the youngest plant among the five? Comparing EFPY's of the 5 plants, aging cannot explain these SCC's by itself. Another possible reason is then, the difference in secondary side environment. Reviewing historical sludge mesurement results of Kori 3&4 and Youngwang 1&2, Younggwang has much more quantity of sludge than that of Kori by about 10 times – 400kg/unit per cycle. In Younggwang 1 and 2, high pressure turbine undercross piping and secondary system piping contains low chromium, which could be the source of FAC(flow accelerated corrosion). Although it is soft sludge that easily removed by secondary side lancing because water chemistry control is being conducted with low calcium to prevent sludge hardening, secondary side data analysis shall be followed more in detail to clarify what caused those cracks.

Conclusions: We've reviewed OD stress corrosion cracks at the hotleg top of tubesheet with thermally treated Alloy 600 tubes found in Korean Model F steam generators. These are just eddy current RPC test results, and neither alternative test nor tube pulling was conducted. It could be easily concluded that it's time to appear the degradation, but why they appeared in just only the youngest steam generator among Korean Model F steam generators is still in question. In the US, only Seabrook and Braidwood have experienced ODSCC with A600TT tubes and Model F steam generators. The root cause turned out to be excessive stress during the cold work process to relief the stress. If it is, Younggwang 2 would be the only plant that stress corrosion crack has initiated without any special reason. It's well known that the most probable cause for differences in OD degradation among similar steam generators is plant specific secondary side environment[6,7]. Historical compare of sludge measurement results shows Younggwang 1&2 have predominantly more sludge than Kori 2, 3&4[3, 4, 8, 9, 10, 11, 12, 13, 14]. It's a kind of soft sludge easily removed by lancing, but secondary side data analysis shall be followed more in detail in the future.

References: [1] KAERI: Root Cause Evaluation Report for Kori 2 Steam Generator Tube Damage – Final Report, May 1995.

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[3] KAITEC: The 13th Younggwang Unit 2 Steam Generator Tube Eddy Current Examination – Final Report, May 2003, p. 41, 44-45 and Appendix 6

[4] KAITEC: The 12th Younggwang Unit 2 Steam Generator Tube Eddy Current Examination – Final Report, November 2001, p. 48 and Appendix 6

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[9] KAITEC: The 12th Kori Unit 3 Steam Generator Tube Eddy Current Examination – Final Report, July 2000, p. 27 and Appendix 6

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[11] KAITEC: The 14th Kori Unit 3 Steam Generator Tube Eddy Current Examination – Final Report, November 2002, p. 30 and Appendix 5

[12] KAITEC: The 12th Kori Unit 4 Steam Generator Tube Eddy Current Examination – Final Report, October 2000, p. 39 and Appendix 5

[13] KAITEC: The 13th Kori Unit 4 Steam Generator Tube Eddy Current Examination – Final Report, February 2002, p. 32 and Appendix 5

[14] KAITEC: The 14th Kori Unit 4 Steam Generator Tube Eddy Current Examination – Final Report, November 2002, p. 32-34 and Appendix 5