Entergy

ENTERGY NUCLEAR NORTHEAST
Engineering Report Cover Sheet

Engineering Report Title:

Operational Assessment of Indian Point 3 Steam Generator Tubing for Cycle 13 and 14

Engineering Report Type:

New ☐ Revision ☒ Cancelled ☐ Superceded ☐

Applicable Site(s)

IP1 ☐ IP2 ☐ IP3 ☒ JAF ☐ PNPS ☐ VY ☐

Quality-Related: ☒ Yes ☐ No

Prepared by: R. Cullen [signature] Date: 03/08/2005
Responsible Engineer

Verified/ Reviewed by: D. C. Ingram D. Curt Ingram Date: 3/8/2005
Design Verifier/Reviewer

Approved by: J. Goldstein [signature] Date: 3/8/2005
Supervisor

Multiple Site Review			
Site	Design Verifier/Reviewer	Supervisor	Date

Operational Assessment of Indian Point 3 Steam Generator Tubing for Cycles 13 and 14

March, 2005

Table of Contents

Revision Summary

Rev.	Description	Changes
0	Original Issue	n/a
1	Revision 1	Incorporates revised steam generator tube structural limits for AVB wear and growth rates for the stretch power uprate operating conditions of 4.8%. Also reviews assumptions for potential degradation given operational experience since revision 0 was prepared.

1. Purpose

The purpose of this report is to provide results from evaluations performed to assess steam generator (SG) tube integrity for next two operating cycles for Indian Point 3 (IP3).

2. Background

Based on the results of Eddy Current (ECT) examinations and analysis described in this report, detection capabilities and industry growth rates, future steam generator tube performance can be evaluated. The intent of this assessment is to evaluate approximately two full-cycles of operation until the next scheduled inspection. Steam generator replacement was performed during the (RF06) refueling outage (June 1989).

A run time of 11.4 EFPY was used to determine the end of cycle conditions. This was based on not inspecting 100% of the tubes during previous outages. In SG 33 and 34, tubes will have gone 7 cycles with out being inspected by RF14. This is from RF08 to RF14 or until the next scheduled inspection.

3. Summary of Results

The steam generators were evaluated to be safe to operate until the next scheduled inspection during (RF14) in the spring of 2007. All performance criteria are anticipated to be maintained with added margin.

4. Evaluation

4.1. Introduction

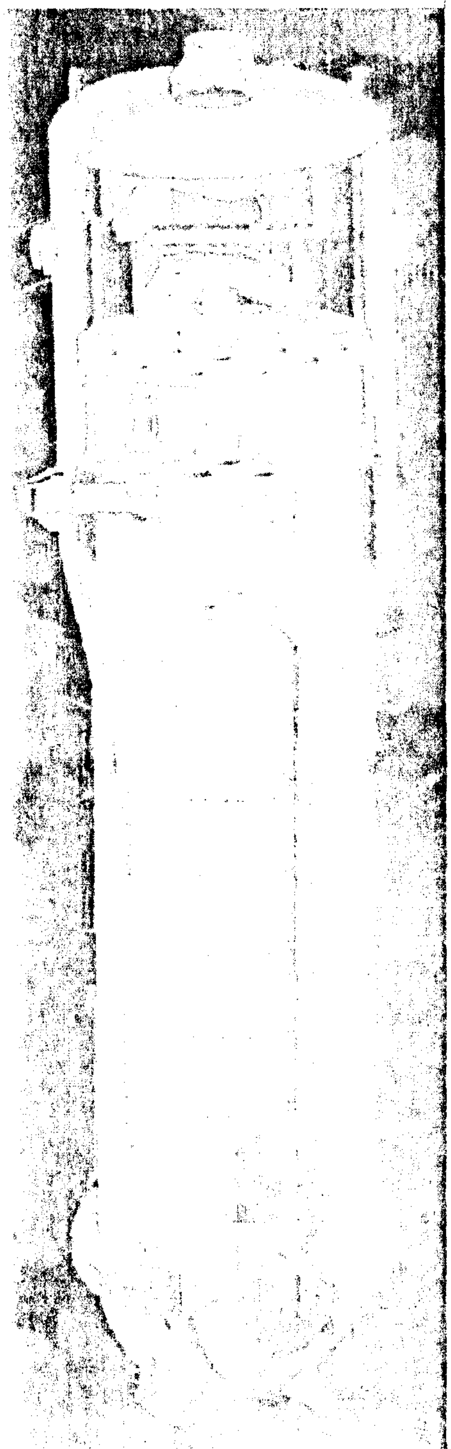
This evaluation follows guidance provided by the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines[1]," for performing condition monitoring and operational assessments of steam generator tubing degradation. Additionally, guidance from the EPRI "Steam Generator Degradation Specific Management Flaw Handbook[2]" was also used.

A steam generator integrity program, which provides reasonable assurance that the steam generator tubes are capable of performing their intended safety function, has been developed by Entergy, using guidance from NEI. This includes establishing performance criteria commensurate with adequate tube integrity, programmatic considerations for providing reasonable assurance that the performance criteria will be met during plant operation, and guidelines for condition monitoring of the SG tubing to confirm that the performance criteria are met.

4.2 Steam Generator Design

[illegible]

Figure 4-1 Westinghouse Model 44F Steam Generator



[BEST COPY]

4.3. Inspection Results

4.3.1. RF12 Scope

The inspection scope for RF12 is listed in Table 4-1 below.

Table 4-1 RF12 Inspection Scope

Inspection	Area/Extent	Inspected	SG	Exams
Bobbin	Full Length of Tube	25%	31	848
		25%	32	852
		25%	33	848
		25%	34	845
Plus Point	U-Bends (Rows 1&2)	60%	31	111
	06H to 06C	60%	32	111
		100%	33	184
		100%	34	187
Plus Point	HL TTS (+/- 3")	20%	31	652
		20%	32	648
		30%	33	965
		30%	34	969
Plus Point	CL TTS (+/- 3")	9%	31	271
		9%	32	270
		9%	33	276
		9%	34	269
Plus Point	HL Dents/Dings	20%	31	10
		20%	32	20
		20%	33	2
		20%	34	9

The historical and planned future bobbin inspections are listed in Table 4-2.

[BEST COPY]

4.3.3. Loose Parts

The design of the moisture separator re-heaters includes moisture separator re-heater demister pads that contain stainless steel wire components. This results in small portions of the wire (1/64" diameter) that migrate to the secondary side of the generator. These items have been identified in previous outages with extensive efforts to find and remove them from the secondary side. There are no cases where this material has caused damage to the tubing. The potential loose part indications appear to be associated with deposits of sludge and/or scale on the outside diameter (OD) of the tubing. The indications are tracked for evaluation in the next inspection. The following is a list of the indications identified in RF12:

Table 4-4 List of Potential Loose Parts Eddy Current Indications

SG	InspDate	Row	Col	Volts	Deg	Ind	Locn	Inch1
31	01-Mar-03	24	46	0.72	77	PLP	TSH	0.09
31	01-Mar-03	29	28	0.27	77	PLP	TSH	2.05
31	01-Mar-03	30	28	0.22	71	PLP	TSH	2.05
34	01-Mar-03	4	45	0.2	70	PLP	TSH	0.49

Visual inspections were performed in all 4 SGs for foreign objects in the annulus and tube lane regions. Additionally, approximately every fifth column was inspected for cleanliness in 33 and 34 SGs following sludge lancing. The presence of any foreign objects seen during the in-bundle inspections was documented as well. The following is a list of the foreign objects identified.

Table 4-5 SG 31 Foreign Object List

Item No.	Foreign Object Location	Foreign Object Description And Dimensions	Retrieved Yes/No
1	CL C: 50 – R: 45	MSR Wire 1/8" x 1/64" Diameter	No
2	CL C: 71 – R: 38	MSR Wire 1/8" x 1/64" Diameter	No
3	CL C: 74 – R: 36	MSR Wire 1/8" x 1/64" Diameter	No
4	CL C: 78 – R: 33	MSR Wire 1/8" x 1/64" Diameter	No
5	CL C: 85 – R: 25	MSR Wire 1/8" x 1/64" Diameter	No
6	CL C: 84 – R: 26	MSR Wire 1/8" x 1/64" Diameter	No
7	TL C: 84 – R: 01	MSR Wire 1/2" x 1/64" Diameter	No
8	HL C: 26 – R: 29	MSR Wire 1/8" x 1/64" Diameter	No

Table 4-6 SG 32 Foreign Object List

No foreign objects were seen in this steam generator.

Table 4-7 SG 33 Foreign Object List

Item No.	Foreign Object Location	Foreign Object Description And Dimensions	Retrieved Yes/No
1	CL C: 46 – R: 45	Flexitallic Gasket 4 3/4" x 1/4" x 1/8"	Yes
2	TL C: 44	MSR Wire 3/8" x 1/64" Diameter	No

Table 4-8 SG 34 Foreign Object List

Item No.	Foreign Object Location	Foreign Object Description And Dimensions	Retrieved Yes/No
1	HL C: 01 – R: 1-5	Wire 3 ¼" x 1/16" Diameter	Yes
2	HL C: 10 – R: 27	MSR wire 1/8" x 1/64" Diameter	No
3	HL C: 21 – R: 38	MSR wire 1/8" x 1/64" Diameter	No
4	HL C: 31 – R: 40	MSR wire ½" x 1/64" Diameter	No
5	HL C: 40 – R: 08	MSR wire 1/8" x 1/64" Diameter	No
6	CL C: 05 – R: 16	MSR wire 1/8" x 1/64" Diameter	No
7	CL C: 31 – R: 35	MSR wire 1/8" x 1/64" Diameter	No
8	CL C: 44 – R: 39	MSR wire 1/8" x 1/64" Diameter	No
9	CL C: 44 – R: 37	MSR wire ¼" x 1/64" Diameter	No
10	CL C: 44 – R: 21	MSR wire ¼" x 1/64" Diameter	No
11	CL C: 74 – R: 33	MSR wire ¼" x 1/64" Diameter	No
12	CL C: 65 – R: 39	MSR wire ¼" x 1/64" Diameter	No
13	CL C: 56 – R: 38	MSR wire ¼" x 1/64" Diameter	No
14	CL C: 56 – R: 37	MSR wire ¼" x 1/64" Diameter	No
15	CL C: 56 – R: 36	MSR wire ¼" x 1/64" Diameter	No
16	CL C: 48 – R: 42	MSR wire ¼" x 1/64" Diameter	No
17	CL C: 48 – R: 38	MSR wire ¼" x 1/64" Diameter	No
18	CL C: 48 – R: 38	MSR wire ¼" x 1/64" Diameter	No
19	CL C: 48 – R: 34	MSR wire 3/8" x 1/64" Diameter	No

4.3.4. Permeability Variation

There were five tubes identified that had permeability variations (PVN). One of these (SG 31 R28:C29) was preventatively plugged due to the magnitude of the interference.

4.3.5. Wear Scars From Sludge Lance Equipment

There were eight tubes identified with wear scars that were attributed to contact with new sludge lance equipment used in the previous outage. One two had indications on both hot and cold legs. The scars were not identified at that time because no eddy current examinations were performed. The cause was consistent with observations made previously at Diablo Canyon and Beaver Valley. Because the wear scars were attributed to maintenance activities no future growth of the degradation is expected, however, the concern over a sizing technique adequate to leave the tubes in service prompted Entergy to administratively plug all tubes found with wear scars.

4.3.6. Repairs

There were a total of 12 tubes repaired during RF12. Three were due to loose part wear from the previous inspection, one due to permeability variation and eight for the newly identified wear scars from the sludge lance equipment. These are listed in Table 4-9 below:

Table 4-9 RF12 Plugging List

SG	Tube	Location	Indication	Percent TW
31	R28 C29	-4.96 to 5.04	PVN	n/a
32	R41 C28	TSH +0.15	VOL	34%
32	R40 C29	TSH +0.0	VOL	32%
32	R41 C29	TSH +0.05	VOL	24%
32	R1 C85	TSH +16.70	VOL	11%*
32	R1 C9	TSC +16.01	VOL	8%*
32	R1 C66	TSC +18.16	VOL	13%*
33	R1 C66	TSH +15.62	VOL	26%*
33	R1 C27	TSH +18.04	VOL	16%*
		TSC +17.86	VOL	12%*
33	R1 C8	TSC +16.51	VOL	9%*
34	R1 C8	TSH +16.69	VOL	10%*
34	R1 C84	TSC +16.92	VOL	11%*

*Sizing results after re-evaluation against ASME flat bottom hole standard documented in reference 13.

4.3.7. In-situ Testing

There were no in-situ pressure tests performed during this examination.

4.3.8. Repair History

Table 4-10 lists the number of the tubes plugged from pre-service to the present. There are no sleeves installed in the IP3 steam generators.

Table 4-10 Tube Repair History - Number of Tubes Plugged

Year	Outage	SG31	SG32	SG33	SG34	Total
1988	Fabrication	0	0	0	2	2
1989	Pre-service	0	0	0	0	0
1990	RF07	0	0	0	0	0
1992	RF08	0	0	0	0	0
1997	RF09	0	0	0	0	0
1999	RF10	0	0	0	0	0
2001	RF11	0	0	0	0	0
2003	RF12	1	6	4	2	13
Totals		1	6	4	4	15

4.4. Operational Assessment

This operational assessment was developed using the deterministic methodology.

4.4.1. Structural Limits

The structural limits for the steam generator tubing were calculated using the methodology outlined in draft Regulatory Guide 1.121. Those limits were updated for the more limiting proposed uprate conditions calculated in 2004 (Reference 14) for operation in cycle 14 and are listed below.

Table 4-11 SG Tube Wear Structural Limits

Location	Parameter	Limit
Straight Leg	t_{min} (inch)	0.024
	Structural Limit (%)	52.0
TSP (1.125")	t_{min} (inch)	0.022
	Structural Limit (%)	55.2
AVB (0.9")	t_{min} (inch)	0.021
	Structural Limit (%)	58.2
AVB (1.50")	t_{min} (inch)	0.024
	Structural Limit (%)	52.0
FDB (0.75")	t_{min} (inch)	0.020
	Structural Limit (%)	61.0

4.4.2. Deterministic Evaluation

There currently are no active degradation mechanisms (as defined by EPRI) in the replacement steam generators. Using industry experience, the only potential issues that the IP3 replacement steam generators would experience is mechanical wear or damage from loose parts. Thermally treated Inconel 690 tubing has been in-service for greater than 10 years at several plants. To date, there has been no degradation identified associated with PWSCC or ODSCC. Mechanical wear will be evaluated assuming detection capabilities and growth for seven cycles of operation. This is the amount of time that a given tube will be in-service without being inspected.

Industry operating experience since revision 0 was prepared in 2003 continues to validate that no corrosion degradation has been found in Inconel 690TT tubing. Primary water stress corrosion cracking (PWSCC) has been found in three plants with Inconel 600TT tubing (Seabrook, Braidwood and Catawba) but laboratory test results indicate that 690TT tubing is immune to the PWSCC phenomenon.

Wear

The following are the inputs used to evaluate AVB wear through the end of Cycle 14:

Method Used	Simplified Statistic
Structural Limit	1.121 analysis (52%TW)
Sizing Uncertainties	Mean of Regression Line + 1.28 sigma
Analyst Uncertainties	1.28 sigma
BOC Flaw Size	Estimate from field and ETSS results
Growth	Value from EPRI [9]

Structural Limit	52 % TW
Sizing Technique Uncertainty[16]	5.74 % TW (90/50)
Analyst Uncertainties[15]	0.86 % TW x 1.28 = 1.1 % TW
BOC Flaw Size	10 % TW
Growth[9] (40 yrs)	12 % TW

The beginning of cycle depth of 10% was estimated based on experience at Indian Point 2 detecting AVB wear at 9% and the dataset for the ETSS had no missed calls all the way down to 5% through wall. No indications of AVB wear have been detected in the Indian Point 3 replacement steam generators from the time of initial service through the last tubing examination in RF12. Indian Point 3 is one of 6 units with Westinghouse model 44F steam generators but the only one with a more advanced AVB design. The design includes 3 sets of bars fabricated of stainless steel that have a contact point twice as long as the other 44F SGs. This means that estimating AVB wear rates based on the experience of the other 44F SGs is overly conservative.

To provide a more realistic estimate of potential AVB wear rates, thermal hydraulic models have been used to predict AVB wear over an assumed 40 year operating life. The initial Westinghouse stress report estimated a maximum wear of 1.3 mils over 40 years. Reference 17 estimates a 40-year post-uprate wear of 2.4 mils or 4.9% TW (through wall).

Reference 9 used a slightly different thermal hydraulic model and estimated a 40-year AVB wear at the most susceptible location to be 6 mils or 12% TW. This assumed a relative high tube to AVB bar clearance of 23 mils when the nominal clearance is about 5 mils resulting in a conservative estimate.

For the purpose of this assessment, the projected AVB growth over the inspection period is assumed to be the 40-year estimate of 12% TW from Reference 9.

To calculate the end of cycle (EOC) maximum depth, the following equation is used:

(BOC flaw) + (SQRT [Sizing^2 + Analyst^2]) + (Growth) = EOC flaw

(10 %) + (SQRT [$5.74^2 + 1.10^2$)] + (12 %) = Maximum Depth at EOC.
(10 %) + (5.8 %) + (12 %) = Maximum Depth at EOC
(10 %) + (5.8 %) + (12 %) = 27.8 %

This result is much less than the lower AVB wear tube structural limit of 52% TW.

Figure 4-3 below was taken from reference 7. This depicts the estimated time to the first wear indication being identified in model 44F Westinghouse generators. Based on the Weibull estimate, Indian Point 3 should find the first indication at ~ 6.35 EFPY. This places it in RF11 outage timeframe in May 2001. This was the outage that an inspection was not performed. In the following outage (RF12), 25 % of all in-service tubes in all four generators were inspected with no wear identified. This is approximately 8.8 EFPY. It is clear that large scale wear is not developing in the Indian Point 3 steam generators.

Figure 4-3 Initiation of AVB Wear in Westinghouse Model 44F Steam Generators

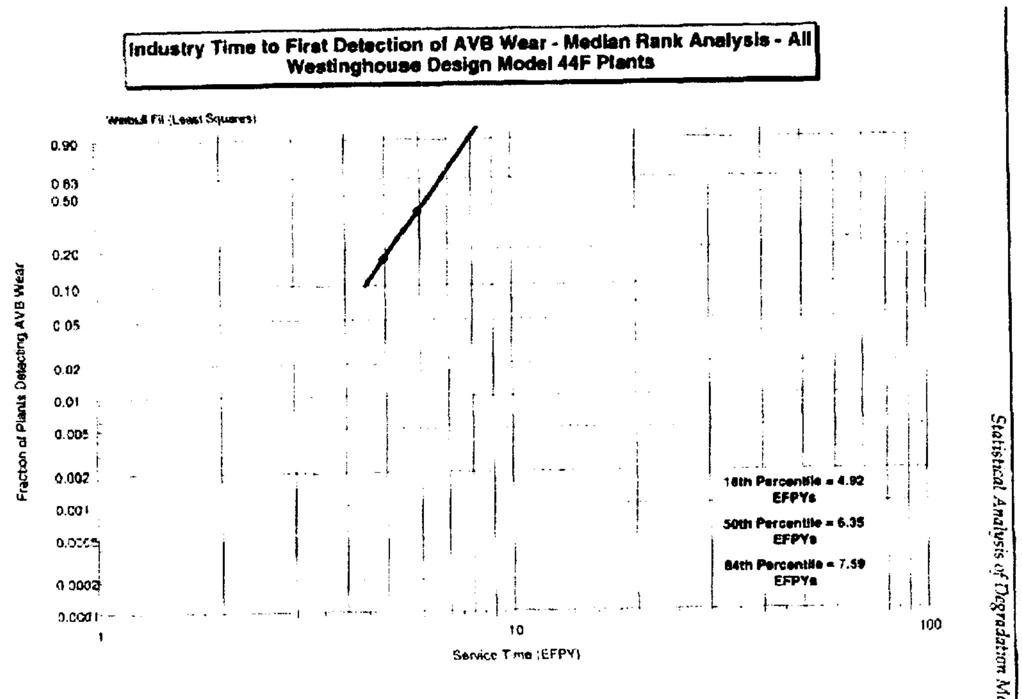


Figure 3-37
Weibull Distribution for Times to First Detection of AVB Wear for Model 44F Plants

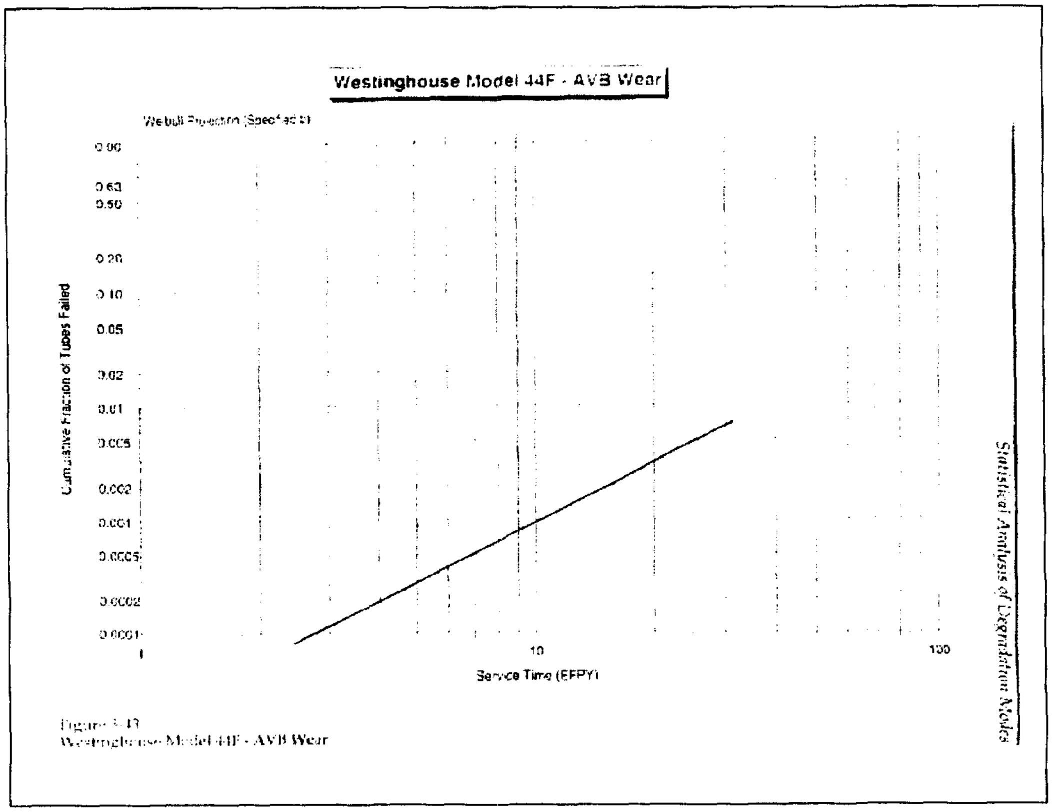
Since wear at the AVBs are typically self limiting, the above calculation should be conservative in nature. Since this value is below the structural limit of 52 %, it is acceptable to operate the plant until RF14 when 100% of the tubing in all four generators will be inspected.

Anticipated Number of Indications

Figure 4-4 was developed by EPRI [7] to determine the initiation of wear in model 44F generators. Based on the next inspection frequency (RF14 or 12.5 EFPY), the anticipated cumulative number of indications should be approximately 0.0015% of the total tubes or:

(0.0015 * 3200) = ~ 5 tubes per generator

Figure 4-4 AVB Wear for Westinghouse Model 44F SG



4.4.3. Loose Parts

As mentioned earlier, there are small portions of the wire (1/64" diameter) that migrate to the secondary side of the generator. These items have been identified in previous outages with extensive efforts to find and remove them from the secondary side. There are no cases where this material has caused damage to the tubing. The potential loose part indications appear to be associated with deposits of sludge and/or scale on the outside diameter (OD) of the tubing. A detailed review of potential loose part intrusion was documented in Westinghouse 00-TR-FSW-024 [11]. The conditions that this evaluation was based on have not changed. The evaluation determined that objects at Indian Point 3 would not have an adverse affect on the generator. It is not anticipated that damage from loose parts should be expected prior to the next inspection in RF14. The indications are tracked for evaluation in the next inspection.

4.4.4. Stress Corrosion Cracking

Many autoclave tests of stressed samples of alloy 600, 690, and 800 tubing materials have demonstrated that alloy 690TT consistently possesses equal or better corrosion and cracking resistance under aggressive environmental conditions than other tubing materials. Recent replacement steam generators utilize tubing composed of 690TT that has been expanded into the steam generator tubesheet by hydraulic methods. AN EPRI [8] study was performed to compare the performance of hydraulically expanded mock-up samples of alloy 690TT tubing to mock-up samples expanded by other methods and containing other alloys under heat transfer conditions representative of recirculating steam generators. Mock-up specimens of thermally treated alloys 600, 690, and 800 with hydraulic expansion, typical of replacement steam generators, outperformed mill annealed alloy 600 with hard rolled or hydraulic expansions. Hard rolled, mill annealed alloy 600 suffered intergranular stress corrosion cracking (IGSCC) between 19 and 30 days. Hydraulically expanded, mill annealed alloy 600 suffered IGSCC between 21 and 132 days of testing. All alloy 690TT mock-ups were resistant to degradation up to and exceeding 132-350 days of exposure with low copper sludge. Based on this study, it is anticipated that Alloy 690 should be a factor of three times at a minimum more resistant than Alloy 600 material. Historically, dependent on variables such as temperature and material properties, units with hard roll transitions have not initiated cracking in ~ 8-10 years of operation. Using this rational, it would be 24-30 years at a minimum before IP3 would see stress corrosion cracking initiated or 2019. Therefore, it is not anticipated that this form of degradation will initiate before the next inspection planned in 2007.

5. Conclusions

Entergy Nuclear Operations has performed an investigation into the potential degradation of the replacement steam generators at IP3. The investigation was based on guidance based on NEI 97-06 for determining end of cycle conditions. The only expected potential condition would be wear based on flow-induced vibration. Using industry experience, it was evaluated that the IP3 steam generators will still meet their structural integrity requirements at the end of 12.5 effective full power years. Therefore, IP3 is considered safe to operate for 2 consecutive cycles without an inspection in the spring of 2005. The next inspection will be during the spring of 2007 (RF14).

This conclusion is still valid after incorporating the potential impact of the stretch power uprate conditions of cycle 14 and industry operating experience since revision 0 was written.

6. References

1. NEI 97-06, "Steam Generator Program Guidelines", Revision 1, January 2001.
2. "Steam Generator Degradation Specific Management Flaw Handbook."
3. WCAP-15920, " Regulatory Guide 1.121 Analysis for the Indian Point Unit 3 Replacement Steam Generators"
4. EPRI ETSS 96004
5. Appendix G Generic NDE Information for Condition Monitoring and Operational Assessments"
6. SG-SGDA-02-42, "Steam Generator Degradation Assessment for Indian Point Unit 3 RFO-12".
7. EPRI TR-108501, "Predicted Tube Degradation for Westinghouse Models D5 and F Type Steam Generators".
8. EPRI TR 104064, "Alloy 690 Qualification: Corrosion Under Prototypic Heat Flux and Temperature Conditions".
9. EPRI 1003145, "Performance Based Steam Generators Inspection Program for Indian Point 3 Nuclear Plant"
10. EPRI TR-107621, "Steam Generator Integrity Assessment Guidelines"
11. 00-TR-FSW-024, Revision 1A,"SG Degradation Assessment for Indian Point 3 R11 Refueling Outage".
12. IP3-RPT-SG-03808, Revision 0, "3R12 Steam Generator Condition Monitoring and Preliminary Operational Assessment Report", April 2003
13. Westinghouse SG-SGDA-03-24, "Re-evaluation of Sludge Lance Rail Wear Scar and Classification of Results", July 2003
14. Westinghouse SGDA-03-147, Revision 1, "Regulatory Guide 1.121 Analysis for Indian Point Unit 3 Replacement Steam Generators for a 4.8% Uprate", April 2004
15. Harris, D.H., "Capabilities of Eddy Current Data Analysts to Detect and Characterize Defects in Steam Generator Tube", Proceedings 15th S/G NDE Workshop, EPRI Report TR107161, November 1996
16. EPRI Eddy Current Examination Technique Specification Sheet, ETSS# 96004.1, Revision 9, February, 2003
17. Westinghouse SGDA-03-124, Revision 0, "The Effect of the Indian Point Unit 3 Total Uprate of 4.8% on Steam Generator Tube Wear", November 2003