TS 6.9.6

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TMI-18-088
September 21, 2018

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Response to Request for Additional Information
TMI Fall 2017 Steam Generator Tube Inspection Report

References:

1. Exelon Letter to the NRC, "TMI-1 Steam Generator Inspection Report for End of Cycle 21 Refueling Outage, dated March 19, 2018 (ADAMS Accession No. ML18085A168)

2. NRC Email to Exelon, "Request for Additional Information Related to TMI Fall 2017 Steam Generator Tube Inspection Report (L-2018-LRO-0014 RAIS), dated August 7, 2018 (ADAMS Accession No. ML18220A783)

By letter (Reference 1) dated March 19, 2018 (Agencywide Documents Access and Management Systems Accession No. ML18085A168), Exelon Generation Company, LLC (Exelon) submitted information summarizing the results of the Fall 2017 Steam Generator (SG) tube inspections performed at Three Mile Island Nuclear Station, Unit 1 (TMI), during refueling outage (RFO) 22. Technical Specification (TS) 6.9.6 requires that a report be submitted within 180 days after the average reactor coolant temperature exceeds 200 degrees Fahrenheit following completion of an inspection performed in accordance with TS 6.19, which requires that an SG Program be established and implemented to ensure SG tube integrity is maintained.

During their technical review of the report, the NRC Staff identified the need for additional information and transmitted the Request for Additional Information (RAI) on August 7, 2018 (Reference 2). The Attachment to this letter provides the response to this RAI.

There are no regulatory commitments contained in this response.

Should you have any questions concerning this response, please contact Mr. Mike Fitzwater at (717) 948-8228.

Respectfully,

EW Callan Jr

E. W. Callan, Jr.
Site Vice President, Three Mile Island, Unit 1
Exelon Generation Company, LLC

Attachment: Response to Request for Additional Information
TMI Fall 2017 Steam Generator Tube Inspection Report

cc:	Regional Administrator - NRC Region I	w/ attachment
	NRC Senior Resident Inspector - TMI	"
	NRC Project Manager, NRR - TMI	"
	R. R. Janati, Pennsylvania Bureau of Radiation Protection	"
	S. L. Martin, PA Department of Environmental Protection, Bureau of Radiation Protection-Nuclear Safety Division	"

ATTACHMENT

Three Mile Island Nuclear Station, Unit 1

Docket No. 50-289

Response to Request for Additional Information
TMI Fall 2017 Steam Generator Tube Inspection Report

Response to NRC Staff's
Request for Additional Information

By letter dated March 19, 2018 (Agencywide Documents Access and Management Systems Accession No. ML18085A168), Exelon Generation Company, LLC (the licensee) submitted information summarizing the results of the fall 2017 steam generator (SG) tube inspections performed at Three Mile Island Nuclear Station, Unit 1, during refueling outage (RFO) 22. Technical Specification (TS) 6.9.6 requires that a report be submitted within 180 days after the average reactor coolant temperature exceeds 200 degrees Fahrenheit following completion of an inspection performed in accordance with TS 6.19, which requires that an SG Program be established and implemented to ensure SG tube integrity is maintained.

In an email dated August 7, 2018, from the NRC (J. Poole) to Exelon (M. Fitzwater, F. Mascitelli, and D. Helker) (ADAMS Accession No. ML18220A783), the NRC provided a final Request for Additional Information (RAI) seeking clarification of certain issues related to the SG tube inspection report, as identified below. In a phone call on August 7, 2018, discussing the draft RAI, Mr. Fitzwater stated that Exelon would provide a response to the RAI question within 45 days (September 21, 2018).

RAI-1 Question

On page 4 of 16, it is stated that "During T1R22 visual and eddy current inspections were performed in the 'B' steam generator only." On the same page, it is later stated that "[d]uring T1R22, all in-service tubes in each steam generator were examined over their entire length using the bobbin coil probe." The NRC staff notes that inspections results were provided only for SG B. Please confirm that inspections were performed in only SG B.

RESPONSE

During T1R22 visual and eddy current inspections were performed in the 'B' steam generator only.

RAI-2 Question

High growth rate flaws have been identified during each of the past several SG inspections. One indication in the tube in Row 49, Column 119 in SG B grew from undetectable in RFO 19 to approximately 63 percent through-wall (TW) by RFO 20. An indication in the tube in Row 2, Column 4 in SG B grew from 28 percent TW in RFO 20 to approximately 73 percent TW by RFO 21.

The NRC staff notes that each of these high growth rate flaws have been identified in SG B and that the root cause for these flaws has been attributed to tube support plate partial locking to the shroud at the wedges and alignment keys. Although SG A has not experienced the same phenomenon with high growth rate flaws, it is unclear to the staff if the same mechanism could become active in SG A at some point during future operation. Please discuss the results of condition monitoring and operational assessment that provide assurance that tube integrity will be maintained for SG A until the next scheduled inspection.

RESPONSE

The mechanism responsible for the anomalously high wear indications observed in SG B over its first four cycles of operation is tube support plate partial locking to the shroud at the wedges and alignment keys during heat-up/operation which deflects the tube supports and alters the normal clearance between the tubes and the broached tube support holes, with the impact being largest on the peripheral tubes.

Since SG B went into service in 2009, this partial locking effect causing high tube wear has consistently been near the "W-axis" alignment key and at several of the tube support plate centering wedges on the periphery. High depth wear on the periphery at higher elevation tube supports, particularly concentrated near the "W-axis", have been observed in all four inspections of SG B performed since replacement. During SG heat-up, interference locally occurs between some tube support plate and shroud locations followed by locking and tube support deflection. The susceptibility to tube supports locking to the shroud is based on the clearances, or lack thereof, between these structures which were created at the time of assembly of the Enhanced Once-Through Steam Generator (EOTSG). Therefore, the clearances in SG B responsible for creating locations susceptible to high wear are not expected to change over time.

Conversely, the SG A is not expected to be subject to the same tube support-to-shroud locking mechanism as observed in SG B over its operational life. This is because if the susceptibility to tube support plate-to-shroud locking had been built into SG A during manufacturing (due to lack of clearance) it would have manifested itself as high wear near the "W-axis" and/or peripheral tubes in the first three inspections performed after SG replacement; which it has not. After three operating cycles (~ 6 years total), this assumption has been confirmed by the observation of the deepest tube support wear reported in SG A in any of the three inspections as only being 33% TW. In addition, SG A has exhibited a more uniform distribution of wear depths as a function of radius from the center of the SG, and unlike SG B, the majority of wear flaws near the W axis in SG A are of the same magnitude as the rest of the population. Therefore, there is reasonable assurance that the tube support-to-shroud locking mechanism affecting SG B will not become active in SG A in the future.

Condition Monitoring assessments have been performed on both SGs since the first inservice inspection (T1R19 in 2011). SG A, unlike SG B, has passed Condition Monitoring acceptance criteria analytically by a wide margin at all three of its inspections, including T1R21. Given that SG A has not exhibited and is not expected to exhibit the high wear behavior as SG B, there is confidence that SG A will again meet Condition Monitoring after operating for two cycles.

While there are, expectedly, certain similarities in the behavior of broached tube support plate wear in both SGs, SG B has consistently exhibited distinct patterns of aggressive wear in certain locations that differ significantly from SG A. SG A lacks distinct patterns of aggressive wear. As a point of comparison, Figure 1 plots the TSP wear depths for SG A and SG B at T1R21 and T1R22, respectively, compared to the T1R22 CM limit curve. This shows the large difference in the depth distribution and margin to the CM limit between the two SGs. In addition, the wear depths and growth rates in SG A are in line with other replacement EOTSGs in the US nuclear fleet. As described above, there is sufficient qualitative and quantitative evidence to support the hypothesis that SG A is not subject to the tube support plate partial locking phenomenon that affects SG B and has been a contributing factor to the aggressive broached tube support plate wear observed. It is therefore justifiable to treat SG A differently than SG B in their Operational Assessments (OAs).

Operational Assessments have been performed with proven fully-probabilistic methods on both SGs since the first in-service inspection (T1R19). Predictions from the OA models have bounded the actual inspection results with respect to wear depths and growth rates, validating the accuracy of the model and method. Additionally, conservatisms are included in the model with respect to the wear growth rates assumed since measured broached tube support plate wear growth rates among all replacement EOTSGs have been observed to attenuate, rather than increase, as operating time increases, as shown in Figure 2 (Upper 95th Percentile Growth Rate) and Figure 3 (Average Growth Rate) below, providing reasonable assurance that wear rates in SG A will follow a similar trend. With respect to the expected size of new TSP wear flaws, a comparison of the maximum new TSP wear depths observed for units with AREVA designed replacement EOTSGs was made. It can be seen in Figure 4 that TMI SG A is trending similarly to SG2, which has justified operating ~4.7 EFPY until the next planned inspection.

With respect to the operational assessment methodology for SG A, the same methods used to justify a single cycle of operation for SG B (T1R22 to T1R23) were applied to SG A and two operating cycles were supported (i.e., T1R21 to T1R23). Specifically, both the Structural Integrity and Accident Induced Leakage Performance Criteria exceeded the minimum required threshold probability of 0.95 at 50% confidence based on SG A operating two cycles. In summary, the methods used to generate the Operational Assessment and their predictions provide sufficient confidence that tube integrity in SG A will be maintained until the next scheduled inspection.

Figure 1: TMI-1 Condition Monitoring of TSP Wear at T1R22 (SG B) and T1R21 (SG A) Based on Bobbin Sizing

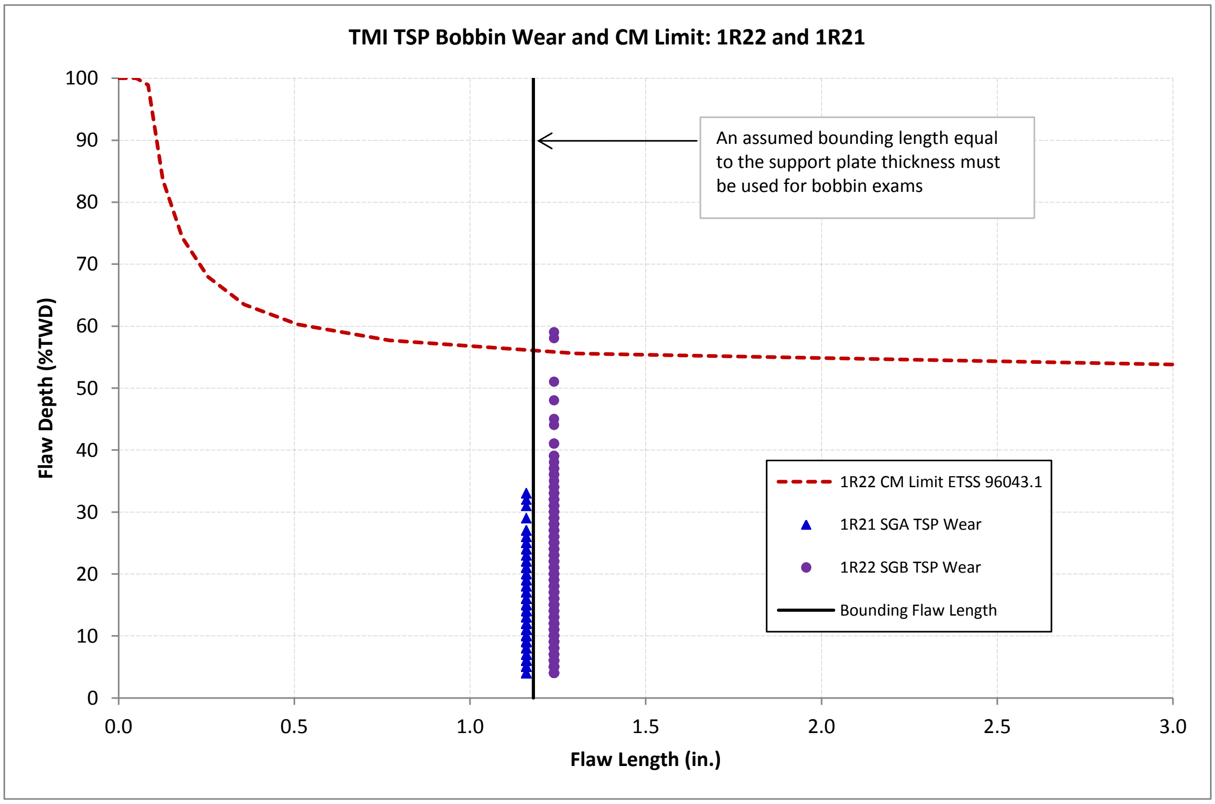

Figure 2: Upper 95th Percentile TSP Wear Growth Rates in Replacement OTSGs

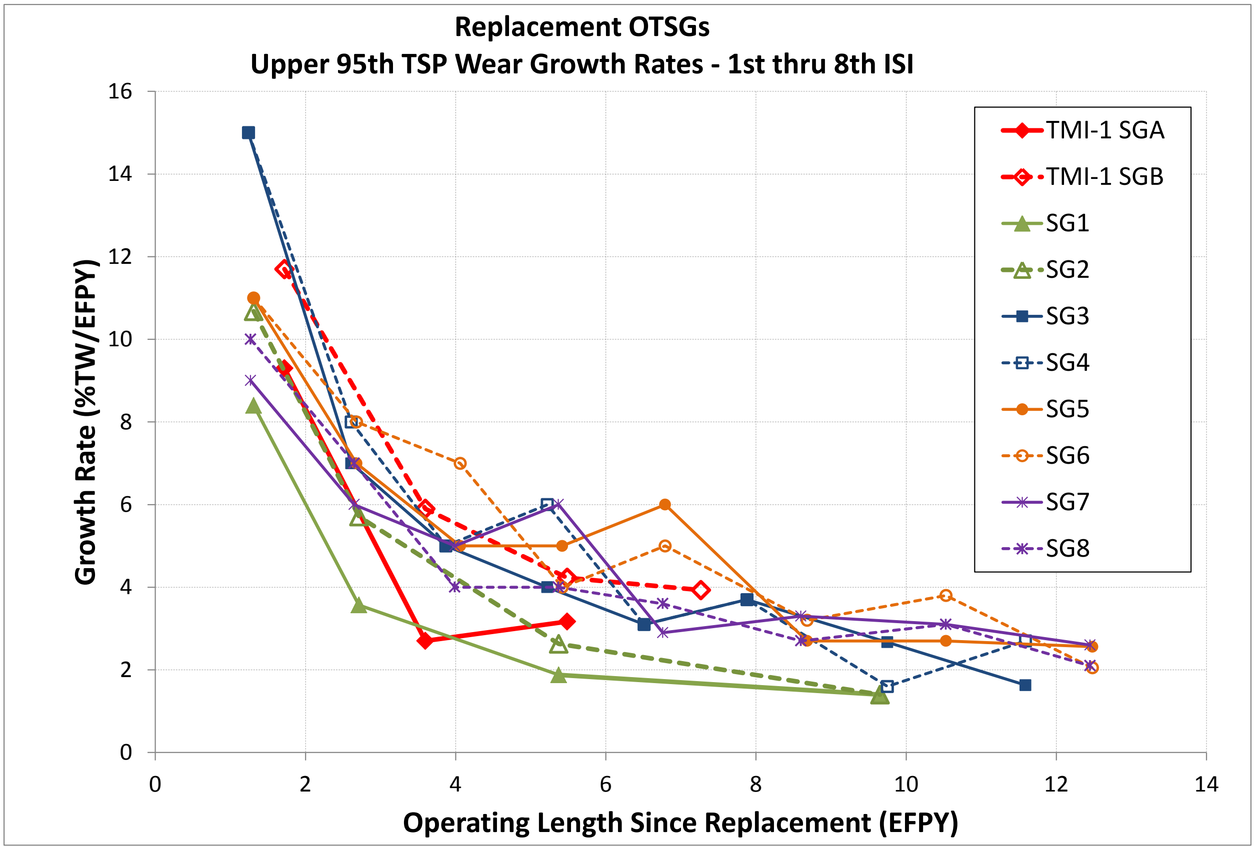

Figure 3: Average TSP Wear Growth Rates in Replacement OTSGs

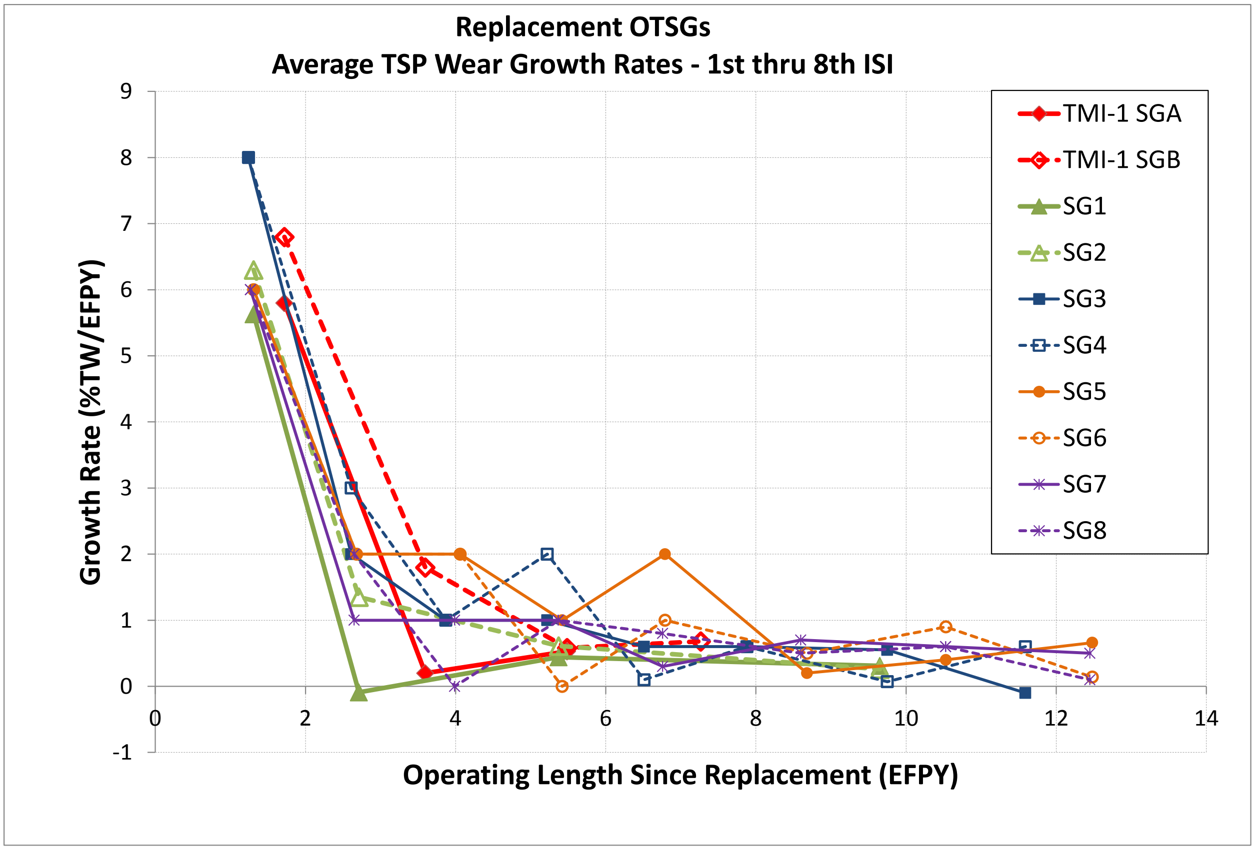

Figure 4: Maximum Depth of New TSP Wear in EOTSGs