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TMI-09-038

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

Subject: Three Mile Island Unit 1 Response to Request for Additional Information Related to Technical Specification Change Request No. 343: Application for Technical Specification Change to Reflect Steam Generator Replacement

References:

(1) AmerGen Letter 5928-08-20010, Three Mile Island, Unit 1, "Technical Specification Change Request No. 343: Application for Technical Specification Change to Reflect Steam Generator Replacement," dated October 9, 2008.

(2) TSTF-449-A, Revision 4, "Steam Generator Tube Integrity."

(3) Letter from P. Bamford (U. S. Nuclear Regulatory Commission) to C. Crane (AmerGen Energy Company, LLC), "Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment 261 Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Generic Letter 2006-01 (TAC Nos. MD1807 and MD0115)", dated September 27, 2007.

(4) Letter from P. Bamford (U. S. Nuclear Regulatory Commission) to C. Pardee (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 -Request for Additional Information Regarding Replacement Steam Generator License Amendment (TAC No. MD9923)," dated March 6, 2009.

By letter dated October 9, 2008 (Reference 1), Exelon Generation Company, LLC (Exelon), formerly AmerGen Energy Company, LLC, requested an Operating License amendment to revise the existing Three Mile Island, Unit 1 (TMI Unit 1) Steam Generator (SG) tube surveillance program. The TMI Unit 1 Technical Specifications (TSs) were previously revised to be consistent with TSTF-449-A, Revision 4 for its current SGs (References 2 and 3). The proposed changes reflect the new thermally treated Alloy 690 tubing design of the replacement SGs and remove sections of the TSs that are not applicable to the replacement SGs.

The U. S Nuclear Regulatory Commission (USNRC) staff has been reviewing the Reference 1 submittal and has determined that additional information is needed to complete its review. The USNRC staff requested additional information on March 6, 2009 (Reference 4).

Exelon's responses to the USNRC questions are provided in the attachment to this letter.

Exelon has determined that the information provided in response to this request for additional information does not impact the conclusions of the No Significant Hazards Consideration as stated in Reference 1.

There are no regulatory commitments contained in this submittal.

A copy of this letter and its attachments are being provided to the designated State Official and the chief executives of the township and county in which the facility is located.

Should you have any questions concerning this letter, please contact Wendy E. Croft at (610) 765-5726.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd day of April, 2009.

Respectfully,

Pamela B. Cowan
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachment Three Mile Island Unit 1 Response to Request for Additional Information Related to Technical Specification Change Request No. 343

cc: Regional Administrator, USNRC Region I
Project Manager, NRR, USNRC – Three Mile Island, Unit 1
Senior Resident Inspector, USNRC – Three Mile Island
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection
Chairman, Board of County Commissioners of Dauphin County, PA
Chairman, Board of Supervisors of Londonderry Township, Dauphin County, PA
R. R. Janati, Commonwealth of Pennsylvania

ATTACHMENT

Three Mile Island Unit 1

Three Mile Island Unit 1 Response to Request for Additional Information Related to Technical Specification Change Request No. 343

Attachment
Three Mile Island Unit 1
Three Mile Island Unit 1 Response to Request for Additional Information
Related to Technical Specification Change Request No. 343

NRC Question 1

Please confirm that the 40 percent through-wall tube repair criterion remains acceptable for the replacement SGs and discuss how it was determined. Specifically, please indicate if it was determined in accordance with Regulatory Guide 1.121, "Bases for Plugging Degraded PWR [pressurized water reactor] Steam Generator Tubes." If the criterion was not established in accordance with RG 1.121, provide the technical basis for the methodology used to develop the tube repair criterion.

TMI Unit 1 Response

An evaluation of the structural and leakage integrity of degraded tubes for the replacement Steam Generators (SGs) at Three Mile Island Unit 1 (TMI Unit 1) has determined that the 40 percent through-wall tube repair criterion remains acceptable for the replacement SGs. The technical basis for the methodology used to develop the repair criteria is NEI 97-06 and the associated Electric Power Research Institute (EPRI) guidelines. The evaluation is performed to the requirements outlined in the latest version of the performance criteria as defined in Section 2.1 of NEI 97-06 "Steam Generator Program Guidelines."

As stated in TSTF–449-A "Steam Generator Tube Integrity":

> "For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as "Steam Generator Degradation Specific Management" (SGDSM). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation-specific approach to managing SG tube integrity has several important benefits. These include:
> - Improved scope and methods for SG inspection,
> - Industry incentive to continue to improve inspection methods, and
> - Development of plugging and repair criteria based on appropriate NDE parameters.
>
> As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated.
>
> Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG Program. These guidelines include:
> - "Steam Generator Examination Guideline"
> - "Steam Generator Integrity Assessment Guidelines"
> - "Steam Generator In-Situ Pressure Test Guideline"
> - "PWR Primary-to-Secondary Leak Guideline"
> - "Primary Water Chemistry Guideline" and
> - "Secondary Water Chemistry Guideline"
>
> These EPRI Guidelines, along with NEI 97-06, tie the entire Steam Generator Program together, while defining a comprehensive, performance based approach to managing SG performance."

Appendix A, "Industry Technical Bases for Structural Integrity Assessment" of EPRI's "Steam Generator Integrity Assessment Guidelines", Revision 2, Document ID 1012987, describes how this method of evaluation is consistent with draft Regulatory Guide 1.121 "Bases for Plugging Degraded PWR [pressurized water reactor] Steam Generator Tubes."

NRC Question 2

For the replacement SGs, please provide the tube size (diameter and wall thickness). Additionally, please specify the design and operating differential pressure across the tubes.

TMI Unit 1 Response

The tubes in the replacement SGs are nominally 0.625" outer diameter with a thickness of 0.0368". Following standard industry practices, the nominal tube dimensions were used for the evaluation of tube structural and leakage integrity.

For the replacement SGs, the limiting normal operating primary-to-secondary differential pressure is 1350 psi; this is the normal operating pressure difference used in the flawed tube repair criterion analysis. Additionally, the design accident condition primary-to-secondary differential pressure is 2575 psi and this value is also used in the flawed tube repair criterion analysis for the replacement SGs.

NRC Question 3

You have proposed to increase your TS limit on primary-to-secondary leakage to 150 gallons per day per SG. In addition, you have indicated that your bases would be modified accordingly. Your current TS 6.19(b)(2) indicates that the safety analysis assumes a leakage volume or rate of primary-to-secondary leakage from all SGs depending on the specific accident analyses. The limit on primary-to-secondary leakage is primarily based on two considerations: (1) limit the frequency of tube rupture and (2) limit the radiological dose to the public and control room operators. Given the current wording in TS 6.19(b)(2), which implies that at least one accident analyses assumes a leakage volume, please confirm that your current NRC-approved design and licensing basis accident analyses that assume primary-to-secondary leakage (e.g., steam line break, locked rotor, control rod ejection, etc.) assume that at least 150 gallons per day per SG primary-to-secondary leakage is occurring. In addition, please provide the reference that incorporated the volume criteria in your design and licensing basis and confirm that this analysis was NRC approved and remains applicable to the new SGs.

TMI Unit 1 Response

The TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) Revision 19 documents three accidents other than steam generator tube rupture that evaluate the consequences of a primary-to-secondary leak. These accidents are the (Control) Rod Ejection Accident, Loss of Electrical Power and (Main) Steam Line Break (MSLB). The current analyses for these three accidents all assume a primary-to-secondary leakage rate of at least 150 gallons per day per SG (300 gallons per day total leakage rate).

The (Control) Rod Ejection and Loss of Electrical Power accidents are described in UFSAR Sections 14.2.2.2 and 14.1.2.8, respectively. Both of these accidents assume a leak rate of 1 gpm for the duration of the event and are unchanged.

The MSLB accident is described in Section 14.1.2.9 of the UFSAR. To support the MSLB accident-induced leakage evaluation of upper tubesheet kinetic expansion indications in the existing SGs, the dose consequences of the volume of primary-to-secondary leakage at integrated average leak rates of greater than 1 gpm were evaluated and then reviewed and approved by the NRC in TMI Unit 1 License Amendment 204, dated October 2, 1997. Technical Specification (TS) 6.19.2 states that leakage from all sources excluding the leakage attributed to the degradation described in TS 6.19.c.1.b is also not to exceed 1 gpm per SG. TS 6.19.c.1.b refers to the MSLB accident-induced leakage evaluation associated with upper tubesheet kinetic expansion indications in the existing SGs; the replacement SGs contain neither upper tubesheet degradation nor kinetic expansion repairs and eliminate the need for accident-induced leakage evaluations of upper tubesheet kinetic expansion indications. The proposed TS change request will restore a 1 gpm per SG leakage limit for the MSLB accident-induced leakage and for all other accidents, other than a steam generator tube rupture. Thus, the accident induced leakage limit for the replacement SGs of 1 gpm as defined in proposed TS 6.19.b.2 is bounded by the existing approved licensing basis for TMI Unit 1 as defined in the licensing basis references cited above.