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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: Summary of Analysis Results Related to PWROG Topical Report BAW-2374, Revision 2.

References:

1. Letter from P. Bamford (U.S. Nuclear Regulatory Commission) to C. Pardee (Exelon Generation Company, LLC), "Three Mile Island, Unit No.1 - Individual Plant Actions Re: Pressurized-Water Reactor Owners Group Topical Report BAW-2374, Revision 2, 'Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping' (TAC NO. ME1797)," dated August 5, 2009

2. Letter from P. Cowan, Exelon Generation Company, LLC, to the U.S. Nuclear Regulatory Commission, "Three Mile Island Unit 1 -Response to June 25, 2009, Public Meeting Between the U.S. Nuclear Regulatory Commission and the Pressurized Water Reactor Owners Group (PWROG) Related to PWROG Topical Report BAW-2374, Revision 2," dated September 4, 2009

By letter dated August 5, 2009 (Reference 1) the U.S. Nuclear Regulatory Commission (USNRC) requested, in part, that Three Mile Island, Unit 1 (TMI, Unit 1) perform an analysis to confirm that the design of the Enhanced Once Through Steam Generators (EOTSGs) is sufficient to withstand the loads associated with a Large Break Loss-of-Coolant Accident (LBLOCA) including the thermal loads associated with a LBLOCA in the Hot Leg Region (HLR) of the Reactor Coolant System (RCS) and to provide the results of that analysis to the USNRC. By letter dated September 4, 2009 (Reference 2) Exelon Generation Company, LLC committed to provide this response to the USNRC by January 31, 2010.

As a result, the TMI, Unit 1 Nuclear Steam Supply System supplier, AREVA, has performed a Thermal Hydraulic Evaluation of thermal loads on the steam generator at 100% power and Hot Zero Power. The results of these analyses are summarized in AREVA Report 51-9125139-001 "Summary Report for Qualification of EOTSG for LBLOCA Loading."

This report concludes that:

- The only portions of the EOTSG significantly affected by the LBLOCA loads are the tube-to-tubesheet weld and tubes with potential wear degradation.
- The Best Estimate and End-of-Cycle (EOC) High Probability Structural Limits for degraded tubing are satisfied for assumed 100% through-wall wear at two lands of the tube support plate (TSP).
- The allowable pop-through depth for a TSP wear scar is 78% through-wall. The ligament pop-through will not lead to a tube "burst".
- The Technical Specification 40% through-wall plugging limit (EOC) has been demonstrated to be conservative and appropriate for circumferential degradation associated with wear type defects.
- The tube-to-tubesheet autogenous weld was parametrically evaluated for the LBLOCA tube load (bounded by the yield load), in accordance with the Level D primary stress requirements of Appendix F of the ASME code, using the strength properties based on Certified Material Test Report data as well as the ASME code. It was concluded that the weld would maintain its structural integrity for the tube axial load resulting from LBLOCA.

Based on the results of the analysis, TMI, Unit 1 confirms that the design of the EOTSGs is sufficient to withstand the loads associated with a LBLOCA including the thermal loads associated with a LBLOCA in the HLR of the RCS.

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

Respectfully,

D. P. Helker

David P. Helker
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Exelon Generation Company, LLC

cc: S. J. Collins, Administrator, Region I, USNRC
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
P. J. Bamford, USNRC Project Manager, TMI Unit 1