UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 3, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ONCE-THROUGH STEAM GENERATOR TUBE LOADS UNDER CONDITIONS RESULTING FROM POSTULATED BREAKS IN REACTOR COOLANT SYSTEM UPPER HOT-LEG LARGE-BORE PIPING (TAC NO. ME1797)

Dear Mr. Pardee:

On June 25, 2009, a public meeting was held between the U.S. Nuclear Regulatory Commission (NRC or Commission) staff and representatives of the Pressurized Water Reactor Owners Group (PWROG) at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, regarding once-through steam generator tube loads under conditions resulting from postulated breaks in reactor coolant system upper hot-leg large-bore piping.

In the meeting, the NRC requested that each Babcock and Wilcox licensee submit a letter summarizing the information discussed at the meeting. The NRC staff followed up this verbal request with a letter dated August 5, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092120053).

Exelon Generation Company, LLC (the licensee) provided a response in its letter dated September 4, 2009 (ADAMS Accession No. ML092470426), for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The NRC staff has reviewed the response and provides the following comments:

- The staff has received your response on the reporting requirements of Section 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," of Title 10 of the *Code of Federal Regulations* (10 CFR). We will contact you if we wish to have further discussions regarding your response.

- Based on its review of this letter, the staff understands that you will be verifying that the design of your replacement steam generators is sufficient to withstand large break loss-of-coolant accident (LBLOCA) loading conditions. In addition, the staff understands that steam generator tube integrity for the replacement steam generators will be maintained for all LBLOCAs (including those in the candy-cane region) as required by Technical Specification (TS) 3.1.1.2, "Steam Generator (SG) Tube Integrity," and TS 6.19, "Steam Generator (SG) Program."

- In your letter dated September 4, 2009, you committed to submit the results of the analysis of LBLOCA loads on the replacement steam generators by January 31, 2010. Subsequent to this submittal, the TMI-1 Updated Final Safety Analysis Report must be updated to reflect this information. Specifically, 10 CFR 50.71(e) requires that licensees update their Final Safety Analysis Report to reflect; (1) information and analyses submitted to the Commission, and/or, (2) evaluations performed in support of conclusions that changes to the facility do not require a license amendment in accordance with 10 CFR 50.59(c)(2).

Based upon the above discussion, the NRC will be closing TAC No. ME1797. If you have any questions, please contact me at 301-415-2833 or by e-mail at Peter.Bamford@nrc.gov.

Sincerely,

Peter Bamford

Peter J. Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

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/ra/

Peter J. Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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