June 7, 2007

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: NRC RECEIPT OF PILGRIM NUCLEAR POWER STATION, RESPONSE TO GENERIC LETTER 2003-01 "CONTROL ROOM HABITABILITY" (TAC NO. MB9840)

Dear Mr. Kansler:

The Nuclear Regulatory Commission (NRC) acknowledges the receipt of your responses to Generic Letter (GL) 2003-01 "Control Room Habitability" dated August 6, 2003 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML032260036); September 30, 2004 (ADAMS Accession No. ML042880297); March 20, 2006 (ADAMS Accession No. ML060890468); March 22, 2007 (ADAMS Accession No. ML070890406); and May 22, 2007 (ADAMS Accession No. ML071510191). This letter provides a status of your response and describes any actions that may be necessary to consider your response to GL 2003-01 complete.

The GL requested that you confirm that your control room meets the design bases (e.g. General Design Criterion (GDC) 1, 3, 4, 5, & 19, draft GDC, or principal design criteria), with special attention to: (1) Determination of the most limiting unfiltered and/or filtered inleakage into the control room and comparison to values used in your design bases for meeting control room operator dose limits from accidents (GL 2003-01, Item 1a); (2) Determination that the most limiting unfiltered inleakage is incorporated into your hazardous chemical assessments (GL 2003-01, Item 1b); and, (3) Determination that reactor control capability is maintained in the control room or at the alternate shutdown location in the event of smoke (GL 2003-01, Item 1b). The GL further requested information on any compensatory measures in use to demonstrate control room habitability, and plans to retire them (GL 2003-01, Item 2).

Entergy reported the results of American Society for Testing Materials (ASTM E741, Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution) tracer gas tests for the Pilgrim Nuclear Power Station control room which is pressurized for accident mitigation. You determined that the maximum tested value for inleakage into the Control Room Envelope (CRE), was 121 (+/- 26) standard cubic feet per minute (scfm), which was more than the value of 10 scfm assumed in your current design basis radiological analyses for Control Room Habitability (CRH). You stated that an operability evaluation of the radiological consequences to control room occupants following a postulated loss-of-coolant accident (LOCA) was developed using alternate source term (AST) methodology and that the evaluation concluded that the dose limits prescribed in GDC-19 are satisfied under worst-case post accident conditions with unfiltered inleakage rates up to 1500 scfm and, therefore, the CRE remains operable and no compensatory actions are required. In your March 20, 2006, letter you stated that you expect to submit a license amendment request (LAR) in the 1st quarter of 2007 to revise your CRH licensing basis using AST methodology.

In your May 22, 2007 letter you stated that the inputs to the original dose consequence calculation were reassessed and that the dose consequence calculation was revised to update inputs with a control room inleakage value of 400 scfm which is greater than the measured inleakage value of 121 +/- 26 scfm. You further stated that the revised calculation demonstrates that the GDC-19 dose limits are met when the Control Room is placed in the isolated mode and that based on the results from the revised dose consequence analysis, you are no longer planning to submit a request to adopt AST methodology.

In response to Item 1(b) of the GL, Entergy indicated that its analysis for hazardous chemicals assumes that the system is operating at full capacity in the purge mode with 100% outside fresh air drawn into the CRE. This analysis is very conservative and in-leakage does not apply. Therefore, no further analysis is needed. You also indicated that reactor control capability is maintained from either the control room or the alternate shutdown panel in the event of smoke.

The GL further requested that you assess your Technical Specifications (TS) to determine if they verify the integrity of the CRE, including ongoing verification of the inleakage assumed in the design basis analysis for CRH, in light of the demonstrated inadequacy of a delta (Δ) P measurement to alone provide such verification (GL 2003-01, Item 1c). In your September 30, 2004, letter you indicated that you plan to use the guidance developed by the Nuclear Energy Institute Technical Specification Task Force (TSTF) and inleakage test results to determine changes to TS if required. In your March 22, 2007, letter you committed to submit a LAR to adopt TSTF-448, Revision 3, by November 30, 2007.

The information you provided supported the fact that there are no compensatory measures needed to be in place to demonstrate CRH, and also supported the conclusion that your are committed to meet the intent of the GDC regarding CRH which is documented in your Updated Final Safety Evaluation Report.

The information as discussed above, and your commitment to submit an LAR based on TSTF-448 is acceptable for purposes of closing out your response to GL 2003-01.

If you have any questions regarding this correspondence, please contact me at 301-415-4125.

Sincerely,

/RA/

James Kim, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-293

cc: See next page

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