July 25, 2006

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601-1839

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PILGRIM NUCLEAR POWER STATION LICENSE RENEWAL APPLICATION (TAC MC9669)

Dear Mr. Kansler:

By letter dated January 25, 2006, Entergy Nuclear Operations, Inc. submitted an application pursuant to 10 CFR Part 54, to renew the operating license for Pilgrim Nuclear Power Station for review by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified, in the enclosure, areas where additional information is needed to complete the review. These relate to Section 2.1 of the LRA on Scoping and Screening Methodology.

These questions were discussed with a member of your staff, Bryan Ford, and a mutually agreeable date for this response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1478 or e-mail RXS2@nrc.gov.

Sincerely,

/RA/

Ram Subbaratnam, Project Manager
License Renewal Branch A
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosure:
Request for Additional Information

cc: See next page

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DATE	7/ 20 /06	7/ 13 /06	7/ 13 /06

OFFICIAL RECORD COPY

Letter to Michael Kansler from R. Subbaratnam dated July 25, 2006:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PILGRIM NUCLEAR POWER PLANT LICENSE RENEWAL APPLICATION (TAC MC9669)

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REQUEST FOR ADDITIONAL INFORMATION
PILGRIM NUCLEAR POWER STATION
LICENSE RENEWAL APPLICATION SECTION 2.5
SECTION 2.1 - SCOPING AND SCREENING METHODOLOGY

1. RAI 2.1-1: Review Methodology for Non-Accident Design Basis Events

10 CFR 54.4(a)(1) states, in part, that systems, structures, and components (SSCs) within the scope of license renewal include safety-related SSCs which are those relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)). 10 CFR 50.49, states that design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed. In regard to identification of design basis events, Section 2.1.3, "Review Procedures," of NUREG-1800 states:

The set of design basis events as defined in the rule is not limited to Chapter 15 (or equivalent) of the [updated final safety analysis report] (UFSAR). Examples of design basis events that may not be described in this chapter include external events, such as floods, storms, earthquakes, tornadoes, or hurricanes, and internal events, such as a high-energy-line break. Information regarding design basis events as defined in 10 CFR 50.49(b)(1) may be found in any chapter of the facility UFSAR, the Commission's regulations, NRC orders, exemptions, or license conditions within the [current licensing basis] (CLB). These sources should also be reviewed to identify systems, structures and components that are relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the functions described in 10 CFR 54.4(a)(1).

During the scoping and screening methodology audit, the NRC staff questioned how non-accident design basis events, particularly design basis events that may not be described in the UFSAR, were considered during scoping. The NRC audit team noted that limiting the review of design bases events to those described in the UFSAR accident analysis could result in omission of safety-related functions described in the CLB.

The staff, therefore, requests the applicant to provide:

a. A list of the design basis events evaluated as part of the license renewal scoping process.

b. A description of the methodology used to ensure that all design basis events (including conditions of normal operation, anticipated operational transients, design basis accidents, external events, and natural phenomena) were addressed during license renewal scoping evaluation.

c. A list of the documentation sources reviewed to ensure that all design basis events were identified.

If, in addressing the above issues, the applicant's review indicates that additional scoping evaluations are required, describe these additional scoping evaluations to address the 10 CFR 54.4(a)(1) criteria. As applicable, list any additional SSCs included within the scope as a result of these efforts, and list those structures and components for which aging management reviews (AMRs) were conducted. For each structure or component describe the aging management programs (AMPs), as applicable, to be credited for managing the identified aging effects.

2. RAI 2.1-2: 10 CFR 54.4(a)(2) Scoping Criteria for Nonsafety-related SSCs

NRC Regulatory Guide 1.188 (Reg. Guide 1.188), "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," Revision 1, dated September 2005, (Reg. Guide 1.188) provided NRC endorsement on the use of NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Revision 6, dated June 2005, (NEI 95-10). Reg. Guide 1.188 indicated that NEI 95 -10, Revision 6, provides methods that the NRC staff considers acceptable for complying with the requirements of 10 CFR Part 54 for preparing a license renewal application (LRA).

NEI 95-10, Appendix F, "Industry Guidance on Revised 54.4(a)(2) Scoping Criterion (Non-Safety Affecting Safety)," (NEI 95-10, Appendix F) discusses non-safety SSCs directly connected to safety-related SSCs. NEI 95-10, Appendix F states, in part, that an equivalent anchor may be defined in the CLB, or may consist of a large piece of plant equipment or series of supports that have been evaluated as a part of a plant-specific piping design analysis. Additionally, the guidance states that an applicant may use a combination of restraints or supports, such that the non-safety piping and associated structures and components attached to safety-related piping, is included in the scope up to a boundary point that encompasses at least two supports in each of three orthogonal directions. The guidance in NEI 95-10, Appendix F also describes as an alternative to identifying a seismic anchor or series of equivalent anchors, the use of bounding criteria which includes using a base-mounted component, a flexible connection, or the free end of the piping run as the end point for the portion of the non-safety piping attached to the safety-related piping to be included in the scope of license renewal.

Section 2.1.1.2.2, "Physical Failure of Nonsafety-related SSCs," of the LRA states the following:

For Pilgrim Nuclear Power Station (PNPS), the "structural boundary" is defined as the portion of a piping system outside the safety class pressure boundary, yet relied upon to provide structural support for the pressure boundary.

Section 2.1.2.1.2, "Identifying Components Subject to Aging Management Review Based on Support of an Intended Function for 10 CFR 54.4.2," of the LRA states the following:

Nonsafety-related piping systems connected to safety-related systems were included up to the structural boundary or to a point that includes an adequate portion of the nonsafety-related piping run to conservatively include the first seismic or equivalent anchor. An equivalent anchor is a combination of hardware or structures that together are equivalent to a seismic anchor. A seismic anchor is defined as hardware or structures that, as required by analysis, physically restrain forces and moments in three orthogonal directions. The physical arrangement as analyzed insures that the stresses that are developed in the safety related piping and supports are within the applicable piping and structural code acceptance limits. This approach included piping beyond the safety/non-safety interface up to a base mounted component, flexible connection, or the end of a piping run (such as a drain line). This is consistent with the guidance in NEI 95-10, Appendix F.

Based on a review of the LRA, the applicant's scoping and screening implementation procedures, and discussions with the applicant, the NRC staff determined that additional information is required with respect to certain aspects of the applicant's evaluation of the 10 CFR 54.4(a)(2) criteria. The staff requests the applicant to provide the following information:

a. Indicate how the structural boundary, which includes the portion of the non-safety piping system outside the safety-related pressure boundary and relied upon to provide structural support for the pressure boundary, was developed. Include a description of the analysis performed to identify the portion of non-safety piping and components required to support the integrity of the safety-related piping and components.

b. Indicate whether equivalent anchors, outside of the analyzed structural boundary and not including the bounding condition terminations (base-component, flexible connection, and end of the piping run), were used. If equivalent anchors, outside of the analyzed structural boundary and not including the bounding condition terminations, were not used, items (c) and (d) below do not need to be addressed.

c. If equivalent anchors, as described in item (b) above, were used, indicate the definition of equivalent anchor which was used for the purpose of the 10 CFR 54.4(a)(2) evaluation and how the definition corresponds to the CLB and to the definition of equivalent anchor listed in NEI 95-10, Appendix F.

d. If equivalent anchors, as described in item (b) above, were used, indicate the number and location of equivalent anchors (i.e., extent of condition).

In addressing each of the above issues, if the review indicates that use of the scoping methodology precluded the identification of any non-safety SSCs that could interact with safety-related SSCs, describe any additional scoping evaluations to be performed to address the 10 CFR 54.4(a)(2) criteria.

As part of your response, list any additional SSCs included within the scope as a result of your efforts, and list those structures and components for which AMRs were conducted. For each structure and component, describe the AMPs, as applicable, to be credited for managing the identified aging effects.

3. RAI 3.0-X: Quality Assurance Program Attributes in Appendix A, "Updated Safety Analysis Report Supplement," and Appendix B, "Aging Management Programs and Activities"

The NRC staff reviewed the applicant's AMPs described in Appendix A, "Updated Safety Analysis Report Supplement," and Appendix B, "Aging Management Programs and Activities," of the LRA, and LRPD-02, "Aging Management Program Evaluation Report," Revision 1. The purpose of this review was to ensure that the quality assurance attributes (corrective action, confirmation process, and administrative controls) were consistent with the staff's guidance described in NUREG-1800, Section A.2, "Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)."

Based on the NRC staff's evaluation, the descriptions of the AMPs and their associated quality attributes provided in Appendix A, Section A.2.1, and Appendix B, Section B.0.3, of the LRA are consistent with the staff's position regarding quality assurance for aging management. However, the description of the corrective action attribute in Section 2.0 of LRPD-02 did not credit the 10 CFR Part 50, Appendix B, quality assurance program.

Therefore, the NRC staff requests that the applicant clarify that the same corrective action program will be applied to all AMPs and that this program meets the requirements of 10 CFR Part 50, Appendix B.