NPA - PD - LR

From: Ram Subbaratnam
To: phb1
Date: 9/8/2006 3:11:39 PM
Subject: Fwd: Final RAI on Sections 3.2.1 and 4.2.1 on the Pilgrim LRA

Here it is !
Ram

>>> Ram Subbaratnam 09/08/2006 3:06 PM >>>
Folks:

Please find attached one set of formal RAIs on section 3.2.1 and 4.2.1 from the NRC staff on the Pilgrim LRA. These are the last set of RAIs on this LRA and hopefully concludes the staff requests. I am sending you an advance copy so that Entergy can work on the lead in responding to the RAIs. As previously agreed upon, you formal response on docket will be due 30 days from the date of receipt of the RAIs in mail.

Thanking you.

Sincerely yours

Ram Subbaratnam
PM Pilgrim LRA
US NRC, (301) 415 1478

Mail Envelope Properties (4501C06B.320 : 19 : 10294)

Subject:	Fwd: Final RAI on Sections 3.2.1 and 4.2.1 on the Pilgrim LRA
Creation Date	9/8/2006 3:11:39 PM
From:	Ram Subbaratnam
Created By:	RXS2@nrc.gov

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September 8, 2006

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

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PILGRIM NUCLEAR POWER STATION LICENSE RENEWAL APPLICATION (TAC NO. MC9669)

Dear Mr. Kansler:

By letter dated January 25, 2006, Entergy Nuclear Operations, Inc., submitted an application pursuant to Title 10 of the *Code of Federal Regulations* 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 and has identified, in the enclosure, areas where additional information is needed to complete the review. These requests for additional information address Sections 3.2 Engineered Safety Features.

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:
As Stated

cc w/encl: See next page

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Office of Nuclear Reactor Regulation

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Enclosure:
As Stated

cc w/encl: See next page

ADAMS Accession No.: ML062500431

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Letter to Michael Kansler from Ram Subbaratnam dated: September 8, 2006

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Kennett Square, PA 19348

REQUESTS FOR ADDITIONAL INFORMATION (RAIs)
PILGRIM NUCLEAR POWER STATION
LICENSE RENEWAL APPLICATION SECTIONS
3.2 ENGINEERED SAFETY FEATURES SYSTEMS

3.2 Engineered Safety Features Systems

RAI 3.2.1 Piping of the Reactor Core Isolation Cooling System is subject to flow accelerated corrosion.

The applicant in LRA Table 3.2.1-19 - engineered safety feature (ESF) and Table 3.2.1 credit the plant-specific Periodic Surveillance and Preventive Maintenance Program for the management of loss of materials (wall thinning). The GALL report recommends using the Flow-Accelerated Corrosion Program, XI.M17, to manage wall thinning. The Periodic Surveillance and Preventive Maintenance Program provides for inspection for material loss (wall thinning) every 5 years but no Monitoring and Trending activities to predict areas of high wall thinning rates or for trending of thinning as does XI.M17. Please provide justification for not providing for monitoring and trending of wall thinning for reactor core isolation cooling (RCIC) piping.

RAI 3.2.2

The applicant in Table 3.2.1 Item 18 of the PNPS LRA stated that none of the ESF system components are within the scope of the BWR Stress Corrosion Cracking Program. Please provide the details that justify this statement.

ENCLOSURE

September 8, 2006

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

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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 by e-mail at 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:
Requests for Additional Information

cc w/encl: See next page

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Enclosure:
Requests for Additional Information

cc w/encl: See next page

ADAMS Accession No.: ML062500401

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DATE	09/08/06	09/08/06	09/08/06

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Letter to Michael Kansler from Ram Subbaratnam dated: September 8, 2006

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REQUESTS FOR ADDITIONAL INFORMATION
PILGRIM LICENSE RENEWAL APPLICATION (LRA)
TIME-LIMITED AGING ANALYSIS (TLAA) OF REACTOR VESSEL (RV) INTERNALS
SECTION: 4.2.1 NEUTRON FLUENCE

RAI 4.2.2-1

Section 4.2.2 of the Pilgrim Nuclear Power Station (PNPS) License Renewal Application (LRA), "Pressure-Temperature [P-T] Limits," states that in a license amendment request dated December 4, 2002, PNPS requested to use the present P-T limit curves through the end of operating cycle 16, which corresponds to approximately 23 effective full power years (EFPY) of facility operation. The end of operating cycle 16 is expected to occur in 2007. Section 4.2.2 also states that, in this December 4, 2002 submittal, PNPS committed to develop and submit updated P-T limit curves and revised fluence calculations based on an NRC-approved calculation method that adheres to Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," prior to the end of operating cycle 16. License Amendment 197 granted this request in 2003. Section 4.2.2 then states that recent fluence calculations that were done per RG 1.190 confirm that the fluence for 54 EFPY is less than the fluence used to calculate the P-T limits that were approved for use only through the end of operating cycle 14. Based on the above statements, you conclude that the current Technical Specification (TS) P-T limit curves remain valid for the period of extended operation.

(a) Please confirm whether the P-T limit curves currently established in the PNPS TSs expire at the end of operating cycle 16 (23 EFPY).

(b) Please explain why the recent fluence calculations that were done per RG 1.190 confirm that the fluence for 54 EFPY is less than the fluence used to calculate the P-T limits that were approved for use through the end of cycle 14. Explain how this information was used to determine that the existing P-T limits remain valid for the period of extended operation.

(c) The staff does not require the P-T limit curves for the extended period of operation to be submitted as part of the applicant's LRA for this time-limited aging analysis (TLAA). However, the staff does require NRC approval of the P-T limit curves for the extended period of operation prior to the expiration of the P-T limit curves for 32 EFPY. Please state when you intend to submit P-T limit curves for NRC approval for the extended licensed period of operation (54 EFPY).

RAI 4.2.4-1

Table 4.2-2 of the PNPS LRA lists initial RT_{NDT} values for the PNPS RV beltline materials. The initial RT_{NDT} values for Lower Intermediate Shell Plate G-3108-1 (Heat No. C-2921-2) and Lower Intermediate Shell Plate G-3108-3 (Heat No. C-2945-2) are less conservative than the corresponding initial RT_{NDT} values established in the NRC staff's reactor vessel integrity database (RVID) for these materials. Section 4.2.4 of the PNPS LRA states that, "initial RT_{NDT} values are from report SIR-00-82, which was submitted in 2001 as part of the PNPS P-T limit change request (Reference 4.2-5)." Reference 4.2-5 points to the April 13, 2001, license amendment issued by the NRC authorizing revised P-T limit curves.

Please provide additional information that points to where the NRC staff authorized the use of the specific initial RT_{NDT} values listed in Table 4.2-4 for determining the adjusted reference temperature (ART) values for the PNPS reactor vessel (RV) beltline materials.

RAI 4.2.4-2

The %Cu and chemistry factor (CF) values for Lower Shell Axial Welds 2-338A, B, and C from LRA Table 4.2-2 are less conservative than the corresponding %Cu and CF values that were established in the staff's RVID for these welds. Please provide the following information:

(a) verification of whether the %Cu and CF values listed in Table 4.2-1 are valid for the above welds,

(b) justification for the use of these chemistry data for the above welds, including the source of the data, and a specific reference for the documentation/analysis demonstrating that these chemistry data represent the best available estimate of the weld chemistries.

RAI 4.2.4-3

Lower Intermediate / Upper Shell Circumferential Weld 3-339B (Heat No. 13253) is listed in the NRC staff's RVID. However this weld is not represented in LRA Table 4.2-2 or (LRA Table 4.2-1). Please resolve this discrepancy.

RAI 4.2.5-1

Section 4.2.5 of the PNPS LRA addresses the TLAA for the RV Circumferential Weld Examination Relief. Table 4.2-3 of the LRA compares the limiting RV circumferential weld parameters for PNPS to those used in the NRC evaluation of the BWRVIP-05 report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations." The PNPS limiting RV circumferential weld parameters are based on Lower Intermediate / Lower Shell Circumferential Weld 1-334 (Heat No. 21935), which is the only circumferential weld represented in LRA Table 4.2-2. However, as discussed in RAI 4.2.4-3, the NRC staff's RVID also lists Lower Intermediate / Upper Shell Circumferential Weld 3-339B (Heat No. 13253) as one of the RV welds for PNPS. Furthermore, the chemistry and CF data for this weld are more limiting than for the Circumferential Weld 1-334. Please explain why this TLAA did not address Lower Intermediate / Upper Shell Circumferential Weld 3-339B (Heat No. 13253).

RAI 4.2.5-2

The NRC staff requires that a request for relief from the American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code) RV circumferential shell weld examination requirements be submitted prior to the beginning of the extended period of operation. Please state whether you intend to apply for relief from the ASME Code RV circumferential weld examination requirements for the extended licensed period of operation. State when you plan to submit this relief request.

RAI 4.2.5-3

In the July 28, 1998 SER on BWRVIP-05, the NRC staff concluded that examination of the RV circumferential shell welds would need to be performed if the corresponding volumetric examinations of the RV axial shell welds revealed the presence of an age-related degradation mechanism. Confirm whether or not previous volumetric examinations of the RV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the welds.

RAI 4.2.6-1

The limiting axial weld failure probability calculated by the NRC staff in the BWRVIP-05 SER is based on the assumption that "essentially 100 percent" (i.e., greater than 90 percent) examination coverage of all RV axial welds can be achieved in accordance with ASME Code, Section XI requirements.

State whether your ISI examinations achieve "essentially 100 percent" (i.e., greater than 90 percent) overall examination coverage for the RV axial welds for the duration of the current licensed operating period. If less than 90 percent overall examination coverage is achieved for the RV axial welds, revise this TLAA to account for the effects of the limited scope examination coverage.