February 23, 2007

Mr. Peter T. Dietrich
Site Vice President
James A. FitzPatrick Nuclear Power Plant
Entergy Nuclear Operation, Inc.
P.O. Box 110
Lycoming, NY 13093-0110

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION REGARDING THE REVIEW OF THE LICENSE RENEWAL APPLICATION FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. MD2666)

Dear Mr. Dietrich:

By letter dated July 31, 2006, Entergy Nuclear Operations, Inc., submitted an application pursuant to Title 10 Code of Federal Regulations Part 54, to be reviewed by the U.S. Nuclear Regulatory Commission (NRC) for renewal of the operating license for James A. FitzPatrick Nuclear Power Plant. The NRC staff performed site audits of the Aging Management Programs and related Time-Limited Aging Analyses provided in the license renewal application. The staff has identified, in the enclosure, an area where additional information is needed to complete the audit reviews.

Based on discussions with Mr. Rick Plasse of your staff, a mutually agreeable date for your response is within 30 days from the date of this letter. If you have any questions regarding this letter or if circumstances result in your need to revise the response date, please contact me at 301-415-1458 or via e-mail at nbl@nrc.gov.

Sincerely,

/RA/

N. B. (Tommy) Le, Senior Project Manager
License Renewal Branch B
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
As stated

cc w/encl: See next page

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Letter to Peter Dietrich, from N B Le dated, February 23, 2007

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REQUESTS FOR ADDITIONAL INFORMATION
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
LICENSE RENEWAL APPLICATION
SECTIONS 4.3.1, 4.7.1, and 4.7.2

Section 4.3.1 Class 1 Fatigue

Background

The staff conducted an on site Time-Limited Aging Analyses (TLAA) follow-up audit on January 8^{th} and 9^{th}, 2007 and has identified the following areas where additional information is required. The following requests for additional information (RAIs) was formerly documented as Question 418 of the staff aging management review database, which has been closed at the request of the U.S. Nuclear Regulatory Commission.

RAI 4.3.1-1

Part A:

Licensing renewal application (LRA) Table 4.3-2 gives the current design basis allowable cycles and updated 60-year cycle projections for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) design basis transients. The cycle values in the "Current Design Basis Cycles, Allowable" column of the table represent the updated current design basis allowable cycles performed by Structural Integrity Associates (SIA) and the cycle values in the "Updated 60 Year Cycle Projection" column of the table represent 60-year cycle projections as of actual JAFNPP operations through Spring 2005. The staff requests the following additional information:

(i). The original current design basis allowable cycles for the original metal fatigue calculations were performed by General Electric Company (GE). Provide the current design basis allowable cycle values that were calculated by GE for the JAFNPP design basis transients.

(ii). Clarify what regulatory process was used to allow SIA's updated current design basis allowable cycle values as the current design basis for the JAFNPP design basis transients.

(iii). Discuss the methods used to establish the original current design basis allowable cycles performed by GE and the updated current design basis allowable cycles by SIA. Identify the differences in the methods used by GE and SIA and justify why SIA's updated current design basis allowable cycle assessment is acceptable to use as the current design basis for JAFNPP.

(iv). For each transient in LRA Table 4.3-2, clarify how many operational cycles have been recorded up to the time that the 60-year transient projections were calculated, as given in the "Updated 60 Year Cycle Projection" column of LRA Table 4.3-2.

(v). Provide a technical discussion to clarify how the 60-year projections were performed based on recorded transient data. In particular, if a particular transient category in LRA Table 4.3-2 is made up of more than one specific transient, clarify which specific transient is used to define the transient and clarify how the total number of cycles were used to derive the 60 year cycle projections.

(vi). Explain how the cycles were recorded prior to 1988, when JAFNPP did not implement a plant computer to track transient events.

(vii). Justify why the following values in LRA Table 4.3-2 are acceptable:

(a). A “Current Design Basis Cycles, Allowable” value of “1" and an “Updated 60 Year Cycle Projection” value of “0" for transient category 13, “Reactor Overpressure.”

(b). A “Current Design Basis Cycles, Allowable” value of “2" and an “Updated 60 Year Cycle Projection” value of “1" for transient category 14, “Single Relief Valve Blowdown.”

(c). A “Current Design Basis Cycles, Allowable” value of “5" and an “Updated 60 Year Cycle Projection” value of “0" for transient category 17, “Improper Start of Cold Recirculation Loop.”

(d). A “Current Design Basis Cycles, Allowable” value of “5" and an “Updated 60 Year Cycle Projection” value of “0" for transient category 18, “Sudden Start of Pump-Cold Recirculation.”

(e). A total “Current Design Basis Cycles, Allowable” value of “233" and a total “Updated 60 Year Cycle Projection” value of “244" for “Shutdowns”, which comprises transient categories Nos. 19, “Reduction to 0% Power;” 20, “Hot Standby;” 21, “Cooldown (100°F/hr to 375°F);” 22, “Vessel Flooding (375°F to 330°F in 10 min.);” and 23, “Cooldown (100°F/hr to 100°F).”

(f). A “Current Design Basis Cycles, Allowable” value of “1" and an “Updated 60 Year Cycle Projection” value of “1" for transient category 24, “Hydrostatic Test (1563 psig).”

(g). A “Current Design Basis Cycles, Allowable” value of “35" and an “Updated 60 Year Cycle Projection” value of “34" for transient category 25, “Unbolt.”

Part B:

Page 19 of GE Design Calculation EAS-149-1286 / DRF B13-01391 discusses GE’s evaluation of 12 transients (i.e., nine reactor SCRAMS, one turbine trip, two feedwater pump trips) that had been grouped into the “Shutdown” transient for the plant. The report stated that the change in reactor coolant temperature (ΔT) for six of these events had exceeded the ΔT value for this transient. The staff noted that the bases provided on page 19 for justifying why these events can be categorized as plant heatups or cooldowns are based on qualitative analysis without using any temperature gradient data. The staff requests the following additional information:

(i). Explain why the six transients specified in GE calculation can be grouped into “Shutdown” transient for the plant when the ΔT values for these six events were determined to be excessive and the temperature gradients for the transients are not defined.

(ii). For the scram event that occurred on November 4, 1984, a ΔT of -297°F and a ΔT of +437°F occurred on the same day, when did ΔT events occur and what were the actual temperature gradients associated with these events.

(iii). Clarify how your response to this part (Part B) factors into your response to Part A, particularly with respect to the number of recorded occurrences for the transient Categories in LRA Table 4.3-2.

Part C:

(i). In the GE stress report, GE characterized 12 unidentified operational transients as reactor SCRAMS. GE identified that 63 occurrences of these transients had occurred prior to 1987.

Verify the operational transients and occurrences identified in the GE stress report and provide your evaluations.

(ii). In LRA Table 4.3-2, Entergy projects that the number of SCRAM events occurring through 60 years of operation for the "All Other SCRAM" events will be 62. Explain how the number of cycles projected through 60 years of operation can be 62 when 63 occurrences had been recorded through 1987.

(iii). In the GE stress report, GE also mentioned that the change in ΔT associated with these 12 unidentified transients was approximately 330°F. The staff requests the following additional information:

(a). Please define these unidentified transients and list the pressure-temperature data for these transients.

(b). Please define the pressure-temperature (P-T) data that were used for the limiting SCRAM event used in SIA's updated 60-year cumulative usage factor calculations.

(c). Justify how these 12 unidentified transients are characterized based on the analyzed P-T limit data used in SIA's updated cumulative usage factor calculations.

(iv). Clarify how your response to this part (Part C) factors into your response to Part A, particularly with respect to the recording the number of cycles for the transients defined in LRA Table 4.3-2 and using this data to project the 60-year cycles for the transients.

Section 4.7.1 - Recirculation Isolation Valves

Background

In qualifying the Recirculation Isolation Valves for TLAA, the Licensee in LRA Section 4.7.1, Recirculation Isolation Valves, partially quotes the method of analysis and qualification requirements stated in Table 16.2-7 of the Updated Final Safety Analysis Report (UFSAR) in reference to the 28" Suction and Discharge Recirculation Valves. It also states that fatigue evaluation was not required for these valves.

The UFSAR indicates that these valves are designed to withstand the effects of cyclic loads and provides the criteria, method and requirements for the fatigue evaluations in Table 16.2-7. Table 16.2-7 provides specific criteria with experimental and analytical methods for evaluating and qualifying the 28" Suction and Discharge Recirculation Valves. For the analytical method it specifies the ASME Boiler and Pressure Vessel Code, Nuclear Vessels Section III Article 4. In

Art 4, Design, fatigue evaluation is part of the required design criteria and the UFSAR in Table 16.2-7 provides cycles with defined service of operation for fatigue evaluations. It also indicates that the results from the fatigue evaluations were plotted and showed that the flange region of the valve is adequate for the defined service. Table 16.2-7 also provides requirements with calculated values and allowables for the primary and the primary plus secondary stresses for the valve body, bonnet and bonnet joint bolts.

Referencing the recirculation isolation valves, the UFSAR states, “For fatigue evaluations consider 30 cycles of normal pressurization followed by blowdown and 270 cycles of normal pressurization followed by normal depressurization.”

As these valves are not ASME class valves, no specific fatigue analysis was required; however, the number of cycles suggested by the UFSAR is greater than the number of cycles allowed as part of the Fatigue Monitoring Program, so the transients suggested will not be exceeded. Thus this TLAA will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(I).

RAI 4.7.1-1

Provide the basis for the referenced cycles in LRA Section 4.7.1 and elaborate on your rational for concluding that the LRA cycles are less than those referenced in the UFSAR Table 16.2-7.

RAI 4.7.1-2

As stated in UFSAR table 16.2-7, fatigue evaluation and interpretation of results was required to qualify the 28" Suction and Discharge Recirculation Valves for 40 year life. Discuss how this UFSAR requirement will be maintained for a 20 year life extension.

RAI 4.7.1-3

UFSAR Table 16.2-7 also provides requirements with calculated values and allowables for the primary and the primary plus secondary stresses for the valve body, bonnet and bonnet joint bolts. In LRA space, determine the suitability of joint bolts for UFSAR defined service cycles projected to end of life extension. If not required provide justification.

RAI 4.7.1-4

LRA Section 4.7.1 states that, “the number of cycles suggested by the UFSAR is greater than the number of cycles allowed as part of the Fatigue Monitoring Program, so the transients suggested will not be exceeded.” Provide a reference that contains the number of cycles allowed in the Fatigue Monitoring Program for the UFSAR defined services for fatigue evaluations stated in UFSAR Table 16.2-7 for the 28" Suction and Discharge Recirculation Valves. In LRA Section 4.7.1, show that these cycles projected to end of life extension are less than the UFSAR cycles.

Section 4.7.2 Leak Before Break

Background

The staff reviewed LRA Section 4.7.2 and per our discussions with the applicant during the staff's and applicant's conference call on 2/12/07, the staff requests that the applicant revise and resubmit Section 4.7.2 after addressing the following points:

1. Remove all references to the terms "Leak Before Break," as the TLAA being discussed clearly predates the formal concept of Leak Before Break documented in Standard Review Plan Section 3.6.3.

2. Rename section 4.7.2 with a title that describes the primary purpose of the analysis in UFSAR Section 16.3.2 and the staff's original Safety Evaluation Report for licensing of the facility, Section 5.2.2.

3. Discuss the specific "time limited" part of the original analysis and provide a description of the "plant modifications" (or non-installations) which were permitted based on the original analysis including locations, or systems in the plant to which the modifications were permitted.

4. Conclude with an explanation of how the TLAA will be addressed in the context of the ongoing LRA review OR include a commitment to be submitted with the completed TLAA for review at least 2 years prior to entering the period of extended operation.

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