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

June 05, 2012

Mr. Michael Perito
Vice President, Site
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE GRAND GULF NUCLEAR STATION LICENSE RENEWAL APPLICATION (TAC NO. ME7493)

Dear Mr. Perito:

By letter dated October 28, 2011, Entergy Operations, Inc., submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54, to renew the operating license for Grand Gulf Nuclear Station, Unit 1 (GGNS) for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The 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 were discussed with Jeff Seiter, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1045 or e-mail nathaniel.ferrer@nrc.gov.

Sincerely,

Nathaniel Ferrer, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosure:
Requests for Additional
Information

cc w/encl: Listserv

GRAND GULF NUCLEAR STATION
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION SET 18

RAI 4.1-1

Background. LRA Table 4.1-2 identifies that the current licensing basis (CLB) does not include a cycle-dependent analysis for intergranular separations (underclad cracking) in reactor pressure vessel (RPV) cladding-to-forging welds. UFSAR Section 5.3.1.2 identifies that the RPV is fabricated primarily from high strength, low-alloy steel plates and forgings. In addition, it identifies that the low-alloy steel RPV forging components were fabricated to SA-508, Class 2 specifications.

Issue. The applicant did not identify which RPV components were fabricated from SA-508, Class 2 forging materials nor provide a discussion in license renewal application (LRA) as to how the underclad cracking issue for SA-508, Class 2 forgings was addressed and resolved in the CLB. The LRA does not include a discussion on how underclad cracking was addressed for those RPV SA-508, Class 2 forging components in the plant design. Thus, the staff does not have sufficient information to determine whether a time-limited aging analysis (TLAA) associated with RPV underclad cracking should be identified in accordance with 10 CFR 54.21(c)(1).

Request.

a. Identify the RPV forging components that were ordered and fabricated to SA-508, Class 2 specifications and whose design included an associated welded cladding.

b. For any RPV forging components identified above, clarify the regulatory process used in the CLB to address underclad cracking in the RPV cladding-to-forging welds.

c. If the CLB resolved the issue of underclad cracking issue by analysis, justify why it does not need to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1).

RAI 4.1-2

Background. LRA Table 4.1-2 identifies that the CLB does not include an analysis for the polar crane that needs to be identified as a TLAA. The applicant stated that the relevant evaluation of the polar crane is not based on time-dependent assumptions defined by the life of the plant; therefore, does not meet the definition of a TLAA in 10 CFR 54.3. UFSAR Section 9 identifies that the facility is designed with the four types of cranes: (1) a polar crane; (2) a containment hatchway crane; (3) a spent-fuel cask crane; and (4) a new fuel handling crane. UFSAR Chapter 9 identifies that the polar crane generally conforms to a CMAA-70 design specification.

Issue. The UFSAR does not specify the design specifications used for the containment hatchway crane, spent-fuel cask crane, or new fuel handling crane. In addition, the staff needs clarification as to whether the design for the containment hatchway, spent-fuel cask, and new fuel handling cranes, and the polar crane established a limit on the number of times the cranes could be used to lift their limiting loads. The staff needs this clarification on the design specifications for these cranes to determine if there is an associated analysis that needs to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1).

Request.

a. Identify the design specifications that were used for the design and analysis of the containment hatchway, spent-fuel cask, and new fuel handling cranes.

b. Clarify whether the design for the containment hatchway, spent-fuel cask, and new fuel handling cranes, and the polar crane established a limit on the number of times the cranes could be used to lift their limiting loads.

c. If the design of these cranes established a limit on the number of crane lifts, justify why these analyses for the cranes do not need to be identified as TLAAs in accordance with 10 CFR 54.21(c)(1).

RAI 4.1-3

Background. SRP-LR Table 4.1-3 identifies that an applicant's CLB may include plant-specific flow-induced vibration analyses for the reactor vessel internal (RVI) components that are TLAAs. LRA Table 4.1-2 identifies that the CLB does not include any flow-induced vibration analysis for the RVI components that would need to be identified as a TLAA. The applicant stated that the flow-induced vibration analyses for the RVI components are not based on time-dependent assumptions defined by the life of the plant and therefore do not conform to the definition of a TLAA in 10 CFR 54.3.

Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," provides an acceptable position that can be used to comply with the design margin requirements for RVI components in 10 CFR 50.34. UFSAR Chapter 3.9.2.3 identifies that the applicant used the results of vibrational analyses from previously operated U.S. nuclear plants ("sister plants") as a baseline and serve as a basis for evaluating the results of the vibrational preoperational and startup tests performed on its own RVI components. UFSAR Section 3A confirms that the applicant used this basis to meet RG 1.20.

Issue. The staff could not determine which "sister plant" assessments (e.g., reports, calculations, or evaluations) are relied upon in the design basis to baseline the results of the applicant's vibrational preoperational and startup tests on the RVI components. In addition, the staff could not determine whether these assessments included analyses of time-dependent vibrational parameters that would need to be identified as TLAAs in accordance with 10 CFR 54.21(c)(1).

Request.

a. Identify all "sister-plant" flow-induced vibrational assessments that are relied upon in the design basis to meet RG 1.20, as discussed in UFSAR Section 3A.

b. For each "sister-plant" assessment identified above, summarize how the assessment analyzed the impacts of flow-induced vibrations on the RVI components and justify why the flow-induced vibrational analysis from the "sister-plant" assessment does not need to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1).

RAI 4.1-4

Background. Paragraph 54.21(c) indicates that license renewal applicants must include a list of TLAAs, as defined in 10 CFR 54.3 and that all identified TLAAs must be dispositioned in accordance with one of the three following acceptance criteria that are specified in 10 CFR 54.21(c)(1)

(i) demonstration that the analysis will remain valid for the period of extended operation

(ii) demonstration that the analysis has been projected to the expiration of the period of extended operation

(iii) demonstration that the effect or effects of aging on the intended function or functions will be adequately managed during the period of extended operation

Issue. During the staff's audit, it was noted in the applicant's basis document for the BWR Feedwater Nozzle Program that a plant-specific fracture mechanics evaluation was performed to support the use of the inspection guidelines in General Electric (GE) NE-523-A71-0594, Revision 1, *Alternate BWR Feedwater Nozzle Inspection Requirements*. The staff noted that this fracture mechanics analysis was based on 40-year projections of the startup/shutdown transients and scram events; however, the applicant did not identify this analysis as a TLAA in accordance with 10 CFR 54.21(c)(1).

Request.

a. Clarify how this plant-specific fracture mechanics evaluation compares to the six criteria for a TLAA as defined in 10 CFR 54.3.

b. Justify whether this fracture mechanics evaluation should be identified as a TLAA in accordance with 10 CFR 54.21(c)(1). If the analysis is a TLAA, provide the necessary revisions to the LRA to support the disposition of the TLAA.

RAI 4.1-5

Background. LRA Section 4.1 identifies that there are no regulatory exemptions that have been approved in accordance with the exemption request acceptance criteria in 10 CFR 50.12 and are based on or predicated on a TLAA.

Issue. The staff noted that Clause 2.D in Operating License No. NPF-29 identifies that the applicant was granted a number of exemptions from the requirements in (1) 10 CFR Part 50, Appendix A, (2) 10 CFR Part 50, Appendix J and (3) 10 CFR Part 100, which were approved based on the exemption request acceptance criteria in 10 CFR 50.12. However, the staff noted that neither the LRA nor the operating license provide sufficient information on the basis for these exemptions or whether the exemptions were based all or in part on an analysis that needs to be identified as a TLAA.

Request. For the exemptions that are still in effect, provide the following information:

a. Summarize in sufficient detail each of the exemptions referenced in Clause 2.D of Operating License, NPF-29, including the basis for requesting the exemption.

b. Clarify whether the basis for requesting these exemptions, prior to their approvals, were based on analyses that would need to be identified as a TLAAs in accordance with 10 CFR 54.21(c)(1).

c. Justify whether the exemptions referenced in Clause 2.D of Operating License, NPF-29, needs to be identified as exemptions that are based on a TLAA and that need to be identified in accordance with 10 CFR 54.21(c)(2).

RAI 4.1-6

Background. LRA Section 1.2 indicates that the licensed thermal power level will be 4408 MWt upon the approval of an extended power uprate (EPU) scheduled to occur in 2012. The applicant indicated that changes to operating parameters due to the increase in power were considered during preparation of the license renewal application, and that a specific example is the higher resulting neutron fluence values used in evaluation of TLAAs in LRA Section 4.2. The applicant also stated that small changes in operating parameters due to the uprated power level have little effect on aging effects requiring management.

Issue. In accordance with 10 CFR 54.3(a), TLAAs are defined as, among other things, calculations and analyses that are contained or incorporated by reference in the CLB. Since the applicant may or may not be approved for operation at EPU conditions by the time the staff anticipates completion of its license renewal review, calculations and analyses in the LRA that reference post-EPU operating conditions and parameters may not be representative of the applicant's CLB.

Request.

a. Identify all the TLAAs discussed in the LRA Section 4 that have utilized post-EPU operating conditions and parameters in their calculations and analyses.

b. Justify how each identified TLAA, which is based on post-EPU operating conditions and parameters, provides an appropriate bounding representation that is conservative for the plant's CLB if the EPU license amendment is not approved prior to the completion of the license renewal review.

RAI 4.2.6-1

Background. LRA Section 4.2.6 addresses RPV core reflood thermal shock by referring to General Electric Report NEDO-10029 described in UFSAR Section 5.3.3. In LRA Section 4.2.6, the applicant indicated that the TLAA for the period of extended operation is based on the analysis results described in the S. Ranganath paper, "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident," Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979. In addition, the operating experience described in LRA Section B.1.11 indicates that during RF14 in 2005, volumetric examinations of the core shroud revealed one indication on the lower side of weld H4 with characteristics associated with stress corrosion cracking (SCC).

<u>Issue</u>. The applicant referred to the Ranganath paper that was not previously included as part of the CLB even though the UFSAR supplement provided in LRA Section A.2.1.6 includes this paper. The staff noted that the LRA includes only a brief evaluation based on adjusted reference temperature (ART) values and temperatures to demonstrate that the S. Ranganath analysis is bounding for the GGNS RPV core reflood thermal shock analysis. However, the LRA does not provide sufficient information for the staff to evaluate for the confirmation of the applicant's claim in the LRA. In addition, the staff needs clarification as to why the LRA does not identify the following items as a TLAA: (1) reflood thermal shock analysis of the RPV core shroud and (2) growth of the core shroud SCC indication.

<u>Request</u>.

a. Provide the following information and calculations to confirm that the GGNS RPV materials are bounded by the analysis in the Ranganath paper as referenced in the LRA.

    1. Additional supporting information and calculations, including the projected fracture toughness values of the RPV materials in comparison to the acceptable RPV fracture toughness values for the postulated event.

    2. Confirm whether the applicant's bounding-case analysis in LRA Section 4.2.6 includes all the relevant RPV materials, the fast fluence values of which are projected to exceed 1E+17 n/cm^2 during the period of extended operation, not omitting any newly identified limiting material or extended beltline material (e.g., not omitting the relevant materials addressed in LRA Table 4.2-2).

    3. Justification that the GGNS RPV is sufficiently similar to the RPV analyzed in the paper for comparison

b. Clarify why the LRA does not identify the reflood thermal shock analysis of the RPV core shroud as a TLAA. As part of the response, if existent, describe how the applicant considered the effect of the observed SCC indication and its growth on the integrity of the core shroud during the reflood thermal shock event.

c. Clarify why the LRA does not identify the growth of the SCC indication in the core shroud as a TLAA. As part of the response, describe whether an analytical evaluation of the flaw has been conducted to project and manage its growth.

RAI 4.7.1-1

<u>Background</u>: LRA Section 4.7.1 refers to UFSAR Section 5.4.4.4, which states that the stainless steel main steam flow restrictors will erode very slowly, and that even with an erosion rate of 0.004 inches per year, the increase in choked flow after 40 years would be no more than five percent. The LRA states that the evaluation of the erosion-corrosion rate for the main steam flow restrictors had determined that the expected erosion-corrosion rate, when operating at velocities at extended power uprate conditions, would be much less than the conservative value of 0.004 inches per year in the UFSAR. The LRA also stated that using this rate, the expected total erosion after 60 years would remain less than the conservative total erosion value identified in the UFSAR for 40 years, and that the analysis had been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(ii).

The staff noted that UFSAR Section 5.4.4, "Main Steam Line Flow Restrictors," provides several design criteria for these components which include (a) limiting the loss of coolant from the vessel to the extent that the reactor vessel level remains high enough to provide cooling within the time required to close the main steam isolation valves and (b) limiting the amount of radiological release outside of the drywell prior to main steam isolation valve closure. The UFSAR also states that "restrictor limits the coolant blowdown rate ...to a maximum (choked) flow of 6.91×10^6 pounds per hour at 1025 psig upstream pressure," and that the design "...limits the steam flow in a severed line less than 170 percent rated flow." The UFSAR further states that a five percent increase in the radiological dose calculation, due to the five percent increase in choked flow rate, is not significant.

Issue: The LRA does not contain information regarding the analysis that demonstrates that the choked flow will remain less than the 170 percent of normal flow or less than 6.91×10^6 pounds per hour at 1025 psig upstream pressure in the event of a main steam line break. In addition, it was unclear to the staff to what value the LRA was referring in the statement "total erosion value identified in the UFSAR for 40 years," since a total erosion value was not given.

The LRA appears to state that the erosion-corrosion rate at extended-power-uprate velocities had been re-evaluated. However, the staff was unable to find any specific information in the correspondence associated with the extended power uprate, to indicate that "the expected erosion-corrosion rate would be much less than the conservative value in the UFSAR."

Request: Provide the results of the projected analysis demonstrating that the intended functions of the main steam flow restrictor are maintained in accordance with the current licensing basis during the period of extended operation. Include the bases for concluding that the expected erosion-corrosion rate would be much less than the conservative value in the UFSAR and the total expected erosion after 60 years would remain less than the conservative total erosion value identified in the UFSAR for 40 years.

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Mr. Michael Perito
Vice President, Site
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE GRAND GULF NUCLEAR STATION LICENSE RENEWAL APPLICATION (TAC NO. ME7493)

Dear Mr. Perito:

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Sincerely,

/RA/

Nathaniel Ferrer, Project Manager
Projects Branch 1
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Office of Nuclear Reactor Regulation

Docket No. 50-416

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