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

June 27, 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 25

RAI 4.2.1-1

Background. LRA Section 4.2.1 addresses the applicant's reactor vessel fluence calculations. LRA Section 4.2.1 states that the fluence is calculated based on a time-limited assumption defined by the operating term, which indicates that the applicant identified the reactor vessel neutron fluence calculations as a time-limited aging analysis (TLAA).

Issue. LRA Table 4.1-1 does not identify the neutron fluence calculation as a TLAA. In addition, the LRA does not address applicant's TLAA disposition of the neutron fluence calculations in terms of the dispositions described in 10 CFR Part 54.21(c)(i), (ii) and (iii).

Request.

a. Clarify why LRA Table 4.1-1 does not identify the neutron fluence calculation as a TLAA.

b. If the fluence calculation is identified as a TLAA, describe the TLAA disposition of the neutron fluence calculation in terms of the dispositions described in 10 CFR Part 54.21(c)(i), (ii) and (iii). Additionally, revise LRA Section 4.2.1, Table 4.1-1 and Section A.2.1.1 to include a relevant TLAA disposition, consistent with the response.

RAI 4.2.1-2

Background. LRA Section 4.2.1 addresses the peak neutron fluence values (E > 1 MeV) for 54 effective full power years (EFPY) based on planned extended power uprate (EPU) power level beginning with Cycle 19. The predicted peak neutron fluence value is 4.44E+18 n/cm^2 at the vessel inner surface of the lower-intermediate shell and axial welds (i.e., Shell Plate 2 location).

The LRA also states that the neutron fluence for the reactor pressure vessel (RPV) beltline region was determined using the General Electric-Hitachi (GEH) method for neutron flux calculation documented in report NEDC-32983P-A and approved by the NRC. The LRA further states the GEH method adheres to the guidance provided in Regulatory Guide (RG) 1.190.

During the audit, the staff noted that Reference 1, which is addressed below, describes the GEH method for applicant's fluence calculations. Reference 1 also refers to Reference 2, which describes another fluence calculation method (MPM method) that the applicant used.

Reference 1: GE Hitachi, Project Task Report, 0000-0104-5984-R0, Revision 0, "Entergy Operations, Inc, Grand Gulf Nuclear Station Extended Power Uprate," Task T0313, RPV Flux Evaluation, October 2009.

Reference 2: MPM-809633, "Grand Gulf Extended Power Uprate Neutron Transport Analysis," August 2009.

In addition, the GEH report referenced above indicates the following information:

(1) The total fluence values at different EFPYs were calculated by adding the corresponding post-EPU fluence to the pre-EPU fluence. The post-EPU fluence and total fluence values are calculated and reported in the GEH report, while the MPM-809633 report calculated the pre-EPU fluence.

(2) Section 3.4.2, "Observations," of the GEH report indicates that calculated post-EPU flux values for core shroud welds H1, V1, V2, V3 and V4 were found to be significantly lower than pre-EPU flux values derived from MPM-809633. This section also states that because the pre-EPU flux values were not calculated by GEH, the reason for this large difference at these locations is unknown.

Issue. Based on the information described above, the staff noted the following concerns.

a. LRA Sections 4.2.1 and A.2.1.1 (UFSAR supplement for reactor vessel fluence) do not identify the methodology described in MPM-809633 as one of the methods that have been used to calculate the neutron fluence. In addition, the staff is not clear as to which fluence calculation methods are included in the current licensing basis.

b. The LRA does not provide information regarding how significantly the post-EPU flux values for shroud welds H1, V1, V2, V3 and V4 based on the GEH methodology are lower than the pre-EPU flux values derived from MPM-809633. The staff is not clear why the post-EPU flux values, which are significantly lower than the pre-EPU flux values, are acceptable.

c. The LRA does not provide information to confirm how the fluence calculation methods of the applicant comply with RG 1.190.

d. The LRA does not describe the results of the measurement benchmarking of the fluence calculation methods with the plant-specific dosimetry data such as the first-cycle or test capsule dosimetry data as addressed in BWRVIP-86, Revision 1, Section 5.4, “Plan for Ongoing Vessel Dosimetry.” It is noted that the staff issued its safety evaluation for BWRVIP-86, Revision 1 by letter dated October 20, 2011.

Request.

a. Justify why LRA Sections 4.2.1 and A.2.1.1 (UFSAR supplement for reactor vessel fluence) do not identify the methodology described in MPM-809633 as one of the methods that have been used to calculate and project the reactor vessel neutron fluence. Alternatively, revise LRA Sections 4.2.1 and A.2.1.1 to identify and include the MPM method in the LRA, as appropriate.

b. As part of the response, clarify what methods for fluence calculations are included in the current licensing basis. If the LRA does not identify all the fluence calculation methods that constitute the current licensing basis, justify why the LRA does not identify all the fluence calculation methods.

c. Provide additional information regarding how the fluence calculation methods have been incorporated into the current licensing basis (e.g., whether through 10 CFR 50.90 or 50.59 process). As part of the response, provide information to demonstrate that such methods are consistent with RG 1.190.

d. Provide the following information related to the flux and fluence calculations.

   1. Provide the pre-EPU (MPM method) and post-EPU (GEH method) flux values for core shroud welds H1, V1, V2, V3 and V4. If existent, describe any other location that involves the higher flux differences between the two calculation methods than these five core shroud welds and provide the associated flux values.

   2. Clarify why the post-EPU flux values (GEH method), which are significantly lower than the pre-EPU flux values (MPM method), are acceptable in terms of the adequacy of the fluence calculations and the compatibility between the fluence calculation methods.

   3. Provide the pre-EPU (MPM method) and post-EPU (GEH method) peak flux values for the inner surfaces of Shell Plates 1 and 2 at the cycle when the EPU is planned to start. If any significant difference exists between these flux values for either of the shell plates, justify why the significant difference is acceptable.

   4. If pre-EPU fluence values obtained using the MPM method were combined with post-EPU fluence values obtained using the GEH method to determine total, post-EPU and end-of-life-extended (EOLE) fluence values, please describe the treatment of uncertainty associated with this technique, and explain how it conforms to the guidance contained in RG 1.190. If the neutron fluence values were combined, and the uncertainty treatment is not believed to adhere to RG 1.190, please justify the acceptability of this approach.

e. Provide additional information to confirm whether the fluence calculation methods have been benchmarked with the ongoing vessel dosimetry, consistent with Section 5.4 of BWRVIP-86, Revision 1. In addition, provide information to confirm whether the fluence calculations using the implemented methods are consistent with the vessel dosimetry data.

RAI 4.7.3-1

Background. In LRA Section 4.7.3, the applicant states that a fluence analysis was performed of components included in the design specification 22A4052 at EPU operating conditions for 60 years plant life. The LRA further states the design specification 22A4052 for the reactor vessel internals components includes requirements beyond the ASME design requirements for austenitic stainless steel base metal components exposed to greater than 1×10^{21} nvt (> 1 MEV) or weld metal greater than 5×10^{20} nvt (> 1 MEV). After location-specific fluence levels were determined, the applicant concludes that the internal core support structure components meet the irradiation criteria in the design specification at EPU operating conditions for 60 years plant life.

Issue. SRP-LR Section 4.7.3.1.2 indicates that for a TLAA disposition pursuant to 10 CFR 54.21(c)(1)(ii), the applicant shall provide a sufficient description of the analysis and document the results of the reanalysis to show that it is satisfactory for the 60-year period. Without this information, the staff cannot evaluate the adequacy of the TLAA.

Request.

a. Justify why only the internal core support structure components were evaluated against the irradiation criteria in the design specification 22A4052.

b. Identify the 40-year fluence levels of the reactor vessel internals components; identify and justify the projected 60-year fluence levels; and identify the design requirements from both the design specification 22A4052 and the ASME code.

RAI B.1.11-1

Background. In Generic Aging Lessons Learned (GALL), Section XI.M9, the scope of the recommended program includes a list of all the Boiling Water Reactor Vessel and Internals Program (BWRVIP) documents that are used as the basis for the BWR Vessel Internals Program.

Issue. In the LRA Updated Final Safety Analysis Report (UFSAR) Supplement Section A.1.11, the applicant stated that "[a]pplicable industry standards and staff-approved BWRVIP documents are used to delineate the program." The applicant also stated that the program is consistent with GALL Report AMP XI.M9. However, during the staff's audit of the UFSAR Supplement for this aging management program (AMP), the staff determined that the UFSAR Supplement summary description did not identify which industry standards and BWRVIP issued documents were within the scope of the BWR Vessel Internals Program.

Request. Modify LRA Section A.1.11 to reference all industry standards and BWRVIP-issued documents that are used to manage the effects of aging for reactor vessel internal (RVI) components that are within the scope of the BWR Vessel Internals Program or justify why LRA Section A.1.11 does not need to identify or reference the subject documents.

Draft RAI B.1.11-2

Background. In the LRA Section B.1.11, "BWR Vessel Internal Program," the applicant stated that "detection of aging effects" program element has been enhanced to manage loss of fracture toughness due to neutron irradiation embrittlement and thermal aging embrittlement for those RVI components that are made from cast austenitic stainless steel (CASS), martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel, or alloy X-750 materials (subject materials). The enhancement states that:

> The susceptibility to neutron or thermal embrittlement for reactor vessel internal components composed of CASS, X-750 alloy, precipitation-hardened (PH) martensitic stainless steel (e.g., 15-5 and 17-4 PH steel), and martensitic stainless steel (e.g., 403, 410,431 steel) will be evaluated. Portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility, neutron fluence, and cracking susceptibility

(i.e., applied stress, operating temperature, and environmental conditions) will be inspected, using an inspection technique capable of detecting the critical flaw size with adequate margin. The critical flaw size will be determined based on the service loading condition and service-degraded material properties. The initial inspection will be performed either prior to or within 5 years after entering the period of extended operation. If cracking is detected after the initial inspection, the frequency of re-inspection will be justified based on fracture toughness properties appropriate for the condition of the component. The sample size will be 100% of the accessible component population, excluding components that may be in compression during normal operations.

Issue. The staff has identified the following issues with the enhancements:

a. In LRA Table 3.1.2-2, only those RVI components made from CASS materials are listed as being managed for loss of fracture toughness. LRA Table 3.1.2-2 does not identify loss of fracture toughness as an aging effect requiring management (AERM) for any RVI components made from martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel or alloy X-750 materials.

b. The enhancement on the BWR Vessel Internals Program does not specify which type of inspection technique or techniques will be used to inspect components manufactured from the subject materials (i.e., CASS, martensitic stainless steel, PH stainless steel, or X-750 nickel alloy). The GALL report identifies that VT-1 visual techniques (including enhanced VT-1 techniques) and volumetric techniques are examples of acceptable inspection methods to detect cracking and indirectly manage loss of fracture toughness in these types of components.

c. It is not evident what criteria will be used to a) select the limiting and expansion components manufactured from the subject materials as being susceptible to thermal and/or neutron embrittlement, b) determine the expansion criteria that triggers expansion of the inspections, and (c) determine the scope of the inspection of the expansion components if expansion is triggered. This issue is consistent with the statement in NUREG-1800, Revision 2 (SRP-LR), Branch Position RLSB-1, which states that provisions on expanding the sample size when degradation is detected in the initial sample should also be included.

d. The enhancement does not define how loss of fracture toughness will be managed in the susceptible, but inaccessible components manufactured from the subject materials when evidence of cracking has been detected in the accessible components.

Request. Based on the points identified in the "Issue" section of this RAI, the staff requests the following information:

a. Are there any components made from martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel or alloy X-750 materials in the BWR vessel internals that are exposed to greater than 1.0E+17 n/cm^2 (E > 1 MeV)? If the answer is yes, then add new line items to the AMR Table 3.1.2-3. If the answer is no, then correct the enhancement to reflect the materials present in the plant.

b. Specify the inspection technique(s) to be used to detect cracking in the enhanced inspections.

c. Discuss the criteria that will be used to determine the components that will be initially inspected. Clearly identify and justify how the lead or limiting susceptible components for loss of fracture toughness due to thermal aging and neutron embrittlement will be chosen. Address the considerations of NUREG-1800, Branch Technical Position RLSB-1, Section A.1.2.3.4, item 4 regarding sampling based condition monitoring plans.

d. Describe how inaccessible components will be addressed:

   1. If determined to be highly susceptible to embrittlement (limiting component).
   2. If not limiting component, but degradation is detected in limiting component.

e. Revise the UFSAR Supplement A.1.11 as necessary to reflect all changes to the enhancement.

RAI B.1.11-3

Background. In GALL, Section XI.M9 for monitoring and trending, the program recommends acceptable documents where additional guidelines for evaluation of crack growth in stainless steels, nickel alloys, and low-alloy steels can be found. During the period of extended operation, core shroud welds and base materials may be exposed to neutron fluence values of 1.0E+21 n/cm^2 (E > 1 MeV) or greater. BWRVIP-100-A report, "Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," provides the BWRVIP's updated fracture toughness data for the irradiated stainless steel materials. For stainless steel materials exposed to neutron fluence equal to or greater than 1.0E+21 n/cm^2 (E >1 MeV), the generic core shroud linear elastic fracture mechanics analyses in Appendix C of the BWRVIP-100-A report used a lower K_{IC} fracture toughness value for the subject materials than the corresponding value reported for these materials in Appendix C of the BWRVIP-76 report, "BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines."

Issue. In LRA Appendix C, the applicant states that BWRVIP-76-A is credited for management of cracking in the Grand Gulf core shroud components. The staff is concerned that, for the "acceptance criteria" activities of stainless steel RVI components, the applicant may not be using the more conservative fracture toughness value for stainless steel materials reported in the BWRVIP-100-A report.

Request. Clarify whether the acceptance criteria for evaluation of cracks in stainless steel RVI components will use the more conservative lower bound fracture toughness value reported for these materials in BWRVIP-100-A. If not, justify the use of a less conservative lower bound fracture toughness value for those RVI components that are made from stainless steel (i.e., use the value reported in BWRVIP-76-A for stainless steel core shroud components or in other applicable NRC-approved BWRVIP reports for other stainless steel RVI components).

RAI B.1.11-4

Background. In BWRVIP-139, the current steam dryer for Grand Gulf is described in detail. During the staff audit of the BWR Vessel Internals Program, the staff determined that a new steam dryer will be installed in May 2012 as part of the applicant's EPU license amendment request.

Issue. The staff is concerned that the applicant may be replacing the Grand Gulf steam dryer with a steam dryer design that is outside of the scope and bounding criteria in the BWRVIP-139-A report.

Request.

1. Provide a description of the new steam dryer that points out the specific differences between the description included in BWRVIP-139 and the actual steam dryer that will be installed as part of the EPU.

2. Identify whether BWRVIP-139-A will continue to be used as the inspection and evaluation (I&E) basis for the new steam dryer. If so, justify why BWRVIP-139-A is considered to be adequate for the inspection and evaluation of the new steam dryers without further augmentation of the BWRVIP's recommended I&E protocols and activities. Otherwise, modify the GGNS LRA Section B.1.11 to include an exception to use an alternative I&E basis in lieu of the BWRVIP-139-A recommendations.

RAI B.1.11-5

Background. In the LRA AMP under operating experience, the applicant has listed inspection history for the current steam dryer at GGNS. There is no mention of operating experience at other plants with similar steam dryers.

Issue. The staff is concerned that the replacement steam dryer at Grand Gulf may develop cracking in the manner that occurred in 2010 for the steam dryers at the Susquehanna nuclear site.

Request. Discuss the relevance of the cracking that occurred in the replacement steam dryers at the Susquehanna nuclear site to the design of the new replacement steam dryers at Grand Gulf. Based on this operating experience (and taking into account the information that will be provided in response to RAI B1.11-4), clarify how the replacement steam dryer and its subcomponents will be inspected at Grand Gulf to detect and manage potential cracking in the steam dryer design during the period of extended operation.

June 27, 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:

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Sincerely,

/RA/

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

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Letter to M. Perito from N. Ferrer dated, June 27, 2012

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE GRAND GULF NUCLEAR STATION, LICENSE RENEWAL APPLICATION

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