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

April 3, 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 Regulation*, 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 1

RAI B.1.6-1

Background. LRA Section B.1.6 states that the Boiling Water Reactor (BWR) Control Rod Drive (CRD) Return Line Nozzle Program is an existing program that manages cracking of the CRD return line nozzle. In comparison, Generic Aging Lessons Learned (GALL) Report AMP XI.M6, "BWR Control Rod Drive Return Line Nozzle," states that the program is a condition monitoring program based on the staff's recommended position in NUREG-0619 for thermal fatigue and the program is also intended to address stress corrosion cracking (SCC) discussed in NRC information notice (IN) 2004-08. IN 2004-08 addresses cracking due to SCC stress of capped CRD return lines.

In addition, the GALL Report AMP XI.M6 "parameters monitored/inspected" program element states that the AMP manages the effects of cracking on the intended function of the reactor vessel, the CRD return line nozzle, and for capped nozzles, the nozzle caps, and cap-to-nozzle welds. GALL Report AMP XI.M6 states that for the volumetric ultrasonic test (UT) examinations that are performed in accordance with this aging management program (AMP), the AMP monitors and evaluates signals that may indicate the presence of a planar flaw (crack).

Issue. During the audit and in interviews with the applicant, the staff noted that the CRD return line was capped to prevent cracking due to cyclic loading. The staff also noted that the applicant's current inservice inspection plan includes the CRD return line nozzle in its scope and sample population. However, the current inservice inspection plan does not have a specific inspection schedule for the capped CRD return line, which indicates that the capped CRD return line is not selected for inspections during the current inservice inspection interval. Therefore, the staff found a concern that the lack of a specific inspection schedule for this component does not ensure adequate detection and management of cracking due to SCC.

Request

a. Provide additional information on the size, material, number of associated welds, and configuration of the capped CRD return line (e.g., 3.5 inches in diameter and Alloy 82/182 nozzle-to-cap butt weld from the low alloy steel nozzle to the Alloy 600 cap).

b. Clarify why the BWR CRD Return Line Nozzle Program is consistent with the GALL Report given the fact that the current inservice inspection plan does not have a specific inspection schedule for the capped CRD return line. As part of the response, clarify if the CRD return line nozzle cap and cap-to-nozzle weld(s) will be examined prior to or during the period of extended operation, and describe the method and schedule of these inspections. Alternatively, provide justification why it is not necessary to examine the CRD return line nozzle cap and cap-to-nozzle weld(s) to detect and manage cracking due to SCC in light of the industry operating experience of cracking due to SCC as described in NRC IN 2004-08.

ENCLOSURE

RAI B.1.7-1

Background. GALL Report AMP XI.M5, "BWR Feedwater Nozzle," recommends enhanced inspections in accordance with General Electric (GE) NE-523-A71-0594, Revision 1, "Alternate BWR Feedwater Nozzle Inspection Requirements."

Issue. During its audit, the staff noted that in the applicant's program basis document for the BWR Feedwater Nozzle Program, the applicant referenced GE-NE-523-A71-0594, Revision 1, consistent with the recommendation in GALL Report AMP XI.M5. However, both LRA Sections B.1.7 and A.1.7 state that the program augments the examinations specified in the ASME Code Section XI, with the recommendation of GE-NE-523-A71-0594. The staff noted that the descriptions in these two sections do not identify the recommended revision of the GE-NE-523-A71-0594 report.

Request. Clarify the revision of the GE-NE-523-A71-0594 report that is used in the BWR Feedwater Nozzle Program and ensure the applicable LRA Sections and Updated Final Safety Analysis Report (UFSAR) supplement accurately reflect this revision. Justify the use of any revision that is not consistent with the recommendation of GALL Report AMP XI.M5.

RAI B.1.8-1

Background. The "parameters monitored/inspected" program element of GALL Report, Rev. 2, AMP XI.M8, "BWR Penetrations," states that the program manages the effects of cracking due to SCC and intergranular stress corrosion cracking (IGSCC) on the intended function of the BWR instrumentation nozzles, CRD housing and incore-monitoring housing (ICMH) penetrations, and BWR standby liquid control (SLC) nozzles/Core ΔP nozzles. The GALL Report also states that the program accomplishes this by inspection for cracks in accordance with the guidelines of approved BWRVIP-49-A, BWRVIP-47-A or BWRVIP-27-A and the requirements of the ASME Code, Section XI, Table IWB 2500-1.

In addition, Section 3.2.5, "Other Inspections," in BWRVIP-47-A indicates that the BWRVIP has determined that removing or dismantling of internal components for the purpose of performing inspections is not warranted to assure safe operation; however, on occasion, utilities may have access to the lower plenum due to maintenance activities not part of normal refueling outage activities. BWRVIP-47-A further states that in such cases, utilities will perform a visual inspection to the extent practical.

Issue. During the audit, the staff noted that in contrast with BWRVIP-47-A, the site documentation for BWR Penetrations Program indicates that the baseline inspections for the CRD housing do not require access to the lower plenum area and currently no additional inspections are recommended beyond the baseline inspections. Additionally, the site documentation shows that if access is gained to the lower plenum (areas below the core plate), accessible areas of the incore flux monitor housing, guide tubes and guide tube stabilizer should be inspected by the VT-3 method. However, it is not clear whether these additional inspections are applied to the incore monitoring housing penetration.

Request.

a. Provide information regarding when the lower plenum housing and penetration are accessible during maintenance activities not part of normal refueling outage activities.

b. Justify why the BWR Penetrations Program does not include the additional inspections of the CRD housing and housing penetration (including stub tubes) described in BWRVIP-47-A, Section 3.2.5. In addition, clarify if the additional inspections are applied to the incore flux monitoring housing penetrations.

c. Ensure the LRA and site documentation are consistent with the response to this RAI.

RAI B.1.8-2

Background. SRP-LR, Table 3.0-1 provides an example of the summary descriptions of aging management programs for the UFSAR Supplement. The summary description for the BWR Penetrations Program in SRP-LR, Table 3.0-1 states that the program includes inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project documents BWRVIP-47-A, BWRVIP-49-A, and BWRVIP-27-A, to ensure the long-term integrity and safe operation of BWR vessel internal components.

The UFSAR supplement in LRA Section A.1.8 states that the BWR Penetrations Program manages cracking of BWR vessel penetrations using inspection and flaw evaluation activities and that applicable industry standards and staff-approved BWRVIP documents are used to delineate the program. In addition, LRA Section B.1.8 states that the BWR Penetrations Program is an existing program that manages cracking of BWR vessel penetrations using inspection and flaw evaluation activities and that applicable industry standards and staff-approved BWRVIP documents are used to delineate the program.

Issue. The applicant's summary description of the BWR Penetrations Program for the UFSAR supplement does not include a specific reference to relevant BWRVIP documents for the program. The UFSAR supplement for the program may not be adequate to ensure program effectiveness due to the omission of specific references to relevant BWRVIP documents for this program. Similarly, LRA Section B.1.8 does not include a specific reference to relevant BWRVIP documents for the program

Request. Justify why LRA Sections A.1.8 and B.1.8 do not include specific references to relevant BWRVIP documents for the BWR Penetrations Program. Alternatively, revise LRA Sections A.1.8 and B.1.8 to include relevant BWRVIP documents for this program.

RAI B.1.8-3

Background. In LRA Appendix C, "Responses to BWRVIP Applicant Action Items Grand Gulf Nuclear Station," the applicant indicated that the core plate differential pressure and standby liquid control (ΔP/SLC) lines inside the reactor vessel have no license renewal intended function and are not subject to aging management review. The applicant also indicated that there is no fatigue time-limited aging analysis (TLAA) applicable to GGNS in BWRVIP-27-A.

During the audit, the staff noted that the site documentation for the Reactor Vessel Internals Program indicates that the design for the ΔP/SLC penetration utilizes an Alloy 600 stub tube set into the bottom head and welded to an Alloy 600 housing. The staff also noted that the ΔP/SLC penetration/nozzle and safe-end assembly forms the reactor coolant pressure boundary (RCPB). In addition, the staff noted that LRA Table 4.3-2, "Cumulative Usage Factors for the Reactor Vessel," includes the 40-year cumulative usage factor for the liquid control-ΔP nozzle, which appears contrary the applicant's claim that there is no fatigue TLAA applicable to GGNS in BWRVIP-27-A.

Issue. In comparison with the applicant's claim that the ΔP/SLC lines inside the reactor vessel have no license renewal intended function, the ΔP/SLC penetration/nozzle and safe-end assembly is part of the reactor coolant pressure boundary. Therefore, it is not clear why no fatigue TLAA is applicable to these components even though this penetration assembly is part of the reactor coolant pressure boundary. In addition, the staff noted that the applicant's claim that no fatigue TLAA is applicable to the ΔP/SLC penetration/nozzle and safe-end assembly is in conflict with the applicant's inclusion of the liquid control-ΔP nozzle in LRA Table 4.3-2 for cumulative usage factors.

Request.

a. Clarify why there is no applicable TLAA for the ΔP/SLC penetration/nozzle and safe-end assembly. As part of the response, resolve the apparent conflict between the applicant's inclusion of the liquid control-ΔP nozzle in LRA Table 4.3-2 and the applicant's claim that there is no TLAA applicable to GGNS in BWRVIP-27-A.

b. Clarify which component is specifically referred to by the "liquid control-ΔP nozzle" among the ΔP/SLC penetration/nozzle and safe-end assembly (e.g., the stub tube and its weld to the penetration). In addition, justify why the "liquid control-ΔP nozzle" is the representative component of the ΔP/SLC penetration/nozzle and safe-end assembly for the fatigue TLAA, and why the other components of the assembly do not need to be addressed in LRA Table 4.3-2.

RAI B.1.9-1

Background. The "scope of program" program element of GALL Report AMP XI.M7, "BWR Stress Corrosion Cracking," states that the program is applicable to all BWR piping and piping welds made of austenitic stainless steel and nickel alloy that are 4 inches or larger in nominal diameter containing reactor coolant at a temperature above 93 °C (200 °F) during power operation, regardless of code classification. The GALL Report recommends the BWR Stress Corrosion Cracking Program to manage cracking due to SCC and IGSCC of stainless steel piping, piping components, and piping elements exposed to treated water greater than 60 °C (140 °F).

LRA Section B.1.9 states that the BWR Stress Corrosion Cracking Program is an existing program that manages cracking of the RCPB using preventive measures, inspection, and flaw evaluation. In addition, LRA 3.2.1-54 indicates that SRP-LR, Table 3.2-1, item 54, was not used and the stainless steel components of the engineered safety features (ESF) subject to evaluation under the BWR Stress Corrosion Cracking Program were reviewed as part of the Class 1 RCPB in LRA Table 3.1.2-3.

Issue. The staff noted that in LRA Table 3.1.2-3 for the RCPB components, the applicant credited the BWR Stress Corrosion Cracking Program to manage the aging effect of RCPB components only. The LRA does not clearly address whether or not the scope of the program includes augmented inspections of non-Class-1 stainless steel piping and its associated welds.

Request. Clarify if the scope of the BWR Stress Corrosion Cracking Program includes non-Class-1 piping and piping welds made of austenitic stainless steel or nickel alloy materials. If the scope of the program does not include non-Class-1 piping and piping welds, justify why non-Class-1 piping and piping welds are excluded from the program scope.

If non-Class 1 piping and welds are within the scope of the BWR Stress Corrosion Cracking Program, revise LRA Sections B.1.9 and A.1.9, as necessary, to clarify that the scope of the program includes the relevant piping and piping welds regardless of code classification.

RAI B.1.9-2

Background. GALL Report AMP XI.M7 states that NUREG-0313, Revision 2 and NRC Generic Letter (GL) 88-01 delineate the guidance for the inspections and the selection of resistant materials and processes that provide resistance to IGSCC. LRA Section B.1.9 states that the BWR Stress Corrosion Cracking Program implements the program delineated in NUREG-0313, Revision 2; GL 88-01; and Supplement 1 of GL 88-01.

Issue. During the audit, the staff noted that the site documentation indicates that the applicant's "detection of aging effects" program element is credited to manage cracking of the following components as well as piping and piping welds: (1) stainless steel thermal sleeves and nickel alloy thermal sleeve extensions of reactor vessel nozzles (recirculation inlet, core spray inlet and RHR/LPCI nozzles), and (2) stainless steel pump casings, valve bodies, and thermowells. However, the inspections recommended in GALL Report AMP XI.M7 and in GL 88-01 apply mainly to piping made of stainless steel or nickel alloy and their associated welds. Therefore, it is not clear what type of inspections will be performed on the thermal sleeves, thermal sleeves extensions, pump casings, valve bodies, and thermowells as part of the BWR Stress Corrosion Cracking Program.

Request. Identify the types of inspections performed on the following components in accordance with the applicant's BWR Stress Corrosion Cracking Program: (1) stainless steel sleeves and nickel alloy thermal sleeve extensions of reactor vessel nozzles (recirculation inlet, core spray inlet, RHR/LPCI nozzles), and (2) stainless steel pump casings, valve bodies, and thermowells. Specifically, identify the inspection methods, sample sizes, and inspection frequencies that will be applied to the inspections of these components during the period of extended operation and justify why inspection methods, sample sizes, and inspection frequencies selected are considered to be capable of detecting and managing cracking in the components during the period of extended operation.

RAI B.1.10-1

Background. LRA Section B.1.10, states that the BWR Vessel Inside Diameter (ID) Attachment Welds Program is an existing program that manages cracking of structural welds for BWR reactor vessel integral attachments. The LRA states that the program is consistent with GALL Report AMP XI.M4, "BWR Vessel ID Attachment Welds," and the inspection of reactor vessel internal attachment welds is governed by the BWRVIP-48-A report. GALL Report AMP XI.M4 indicates that the program includes inspection and flaw evaluation in accordance with the guidelines in the BWRVIP-48-A report.

Issue. The UFSAR Supplement in LRA Section A.1.10 states that applicable industry standards and staff-approved BWRVIP documents are used to delineate the program. The staff noted that the summary description for this program in the UFSAR Supplement does not identify the use of the BWRVIP-48-A report, which is recommended by GALL Report AMP XI.M4.

Request. Justify why the BWRVIP-48-A report does not need to be identified in LRA Section A.1.10, or revise LRA Section A.1.10 to indicate that the BWR Vessel ID Attachment Welds Program uses inspections and flaw evaluation in accordance with the guidelines in the BWRVIP-48-A report.

RAI B.1.23-1

Background. LRA Section B.1.23, "Inservice Inspection," states that, "ISI Program Summary Reports between 2004 and 2010 reveal compliance and provide evidence that the program is effective for managing aging effects in accordance with the ASME Boiler Pressure Vessel Code Section XI."

GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," states that the program "has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants." It also provides industry operating experience cases in the "operating experience" program element.

Issue. The "operating experience" program element of the Inservice Inspection (ISI) Program indicates that the program is consistent with the GALL Report and is in compliance with the ASME Code. However, the LRA does not provide any detailed discussion, beyond the general statement about the Code compliance, to demonstrate the effectiveness of the program in detecting and managing the aging effects of Class 1, 2, or 3 components. The applicant's ISI Program Summary Reports between 2004 and 2010 provide brief inspection results to establish the program's compliance with the ASME Code, but do not provide any discussion to demonstrate program effectiveness in the context of monitoring, detecting, and correcting aging degradation. It is not clear how the applicant's plant-specific operating experience demonstrates the effectiveness of the program (e.g., detection of aging effects and directing corrective actions in a timely manner).

Request.

a. Provide detailed representative operating experience related to the Inservice Inspection Program in order to demonstrate that the program is effective in managing aging effects in accordance with ASME Code Section XI. The discussion should provide context for any specific example in terms of (1) detection of aging effects such as indications of cracking or loss of material, (2) monitoring and trending of aging effects such as results of flaw evaluation and subsequent inspections, and (3) timely corrective actions such as inspection sample expansion and repair/replacement activities.

b. If a need is identified for expanding the program elements based on the plant-specific operating experience review, enhance the program accordingly. In addition, revise the applicable LRA sections, as necessary, consistent with this RAI response.

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

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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 Regulation*, 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.

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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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