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GNRO-2012/00085

July 25, 2012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Response to Request for Additional Information (RAI) Set 25 dated June 27, 2012
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCE: NRC Letter, "Requests for Additional Information for the Review of the Grand Gulf Nuclear Station License Renewal Application," dated June 27, 2012 (GNRI-2012/00143)

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in the Attachment, the response to the referenced Request for Additional Information (RAI).

This letter contains no new commitments. If you have any questions or require additional information, please contact Christina L. Perino at 601-437-6299.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of July, 2012.

Sincerely,

MP/JAS

Attachment: Response to Request for Additional Information (RAI)

cc: with Attachment

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Attachment to

GNRO-2012/00085

Response to Request for Additional Information (RAI)

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 4.2.1-1 RESPONSE

a. The neutron fluence calculation is not a time-limited aging analysis (TLAA) since, as a stand-alone analysis, it does not meet the definition in 10 CFR 54.3(a). Specifically, a neutron fluence calculation does not "consider the effects of aging," which is the second element of the six-element definition of a TLAA in 10 CFR 54.3(a). The neutron fluence calculation results are used as inputs into fracture toughness analyses that consider the effects of aging due to exposure to neutron irradiation. Those analyses are appropriately evaluated as TLAAs as described in License Renewal Application (LRA) Section 4.2.
b. Although the neutron fluence calculation does not meet the 10 CFR 54.3(a) definition of a TLAA, the calculation is projected to the end of the period of extended operation for use in demonstrating that neutron embrittlement analyses have been projected to the end of the period of extended operation. The validity of the neutron embrittlement TLAAs is based on a valid input of neutron fluence at the end of the period of extended operation.

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 4.2.1-2 RESPONSE

a. Reactor vessel neutron fluence for extended power uprate (EPU) operating conditions, through the period of extended operation, is calculated in accordance with *Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation, NEDC-32983P-A, Rev. 2, January 2006.* Pre-EPU fluence values were generated from the MPM Technologies, Inc. (MPM) analysis. MPM analysis is consistent with the guidance contained in RG 1.190 and is approved by the NRC in *TAC No. MB6687, Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment RE: Pressure-Temperature Limit Curves and Tables, October 27 2003*, as described in UFSAR Section 4.3.2.8.

LRA Sections 4.2.1 and A.2.1.1 are revised as shown below to include discussion of the MPM fluence method.

b. The two fluence calculation methods included in the current licensing basis are 1)the post-EPU General Electric-Hitachi (GEH) method which is reviewed and approved in General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation, NEDC-32983P-A, Rev. 2, January 2006, and 2) the pre-EPU MPM method which is reviewed and approved by the NRC in *TAC No. MB6687, Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment RE: Pressure-Temperature Limit Curves and Tables, October 27 2003*, as described in UFSAR Section 4.3.2.8. LRA Sections 4.2.1 and A.2.1.1 are revised below to include discussion of the MPM fluence method.

c. The post-EPU GEH fluence method was incorporated into the current licensing basis through the 10 CFR 50.90 process during EPU license amendment approval and is consistent with Regulatory Guide 1.190 as described in NEDC-32983P-A, Rev. 2. The pre-EPU MPM fluence method was incorporated into the current licensing basis through the 10 CFR 50.59 process and is also consistent with Regulatory Guide 1.190 as described in UFSAR Section 4.3.2.8.

d.1 **Maximum Fast Flux at GGNS Top Guide Welds**

Weld Identification	**Pre-EPU Peak Flux (n/cm^2-s)**	**Post-EPU Peak Flux (n/cm^2-s)**
H1	1.04E10	1.23E07
V1	1.35E10	1.02E07
V2	1.35E10	9.64E06
V3	1.35E10	1.02E07
V4	1.35E10	9.64E06

Welds H1, V1, V2, V3, and V4 are welds on the Grand Gulf Nuclear Station (GGNS) top guide that sits above the core shroud. No locations evaluated in the post-EPU GEH fluence evaluation except for H1, V1, V2, V3, and V4 were found to have flux values lower than pre-EPU flux values. The apparent discrepancy between the pre-EPU flux values and the post-EPU flux values has been entered into the GGNS corrective action program.

d.2 Post-EPU (GEH method) flux values are acceptable since the method of their calculation is the NRC-approved fluence calculation method of licensing topical report NEDC-32983P-A, Rev. 2. With the acceptability of the post-EPU flux values, combining these with higher pre-EPU flux values obtained from a different method (MPM method) is conservative. The apparent discrepancy between the pre-EPU flux values and the post-EPU flux values has been entered into the GGNS corrective action program. The pre-EPU values appear overly conservative as they are associated with a location that is approximately 45 inches above the top of the active fuel where flux values are expected to be lower than values at the top of the active fuel. Even considering the apparently conservative flux values, the resulting fluence at this location is significantly less than the peak fluence on the core shroud welds.

d.3 Pre-EPU (MPM method) peak flux for the reactor pressure vessel (RPV) inner surface is 1.71E+9 n/cm^2-s. Post-EPU (GEH method) peak flux for the RPV inner surface is 2.90E+9 n/cm^2-s. The ratio of post-EPU to pre-EPU peak reactor vessel ID flux is 1.69. Some of the increase in flux level is due to the ~15% thermal power increase from power uprate. Additional causes include higher power in peripheral bundles due to EPU operating conditions. However, both pre-EPU (MPM) and post-EPU (GEH) methods are consistent with Regulatory Guide 1.190 and have been approved by the NRC as described in UFSAR Section 4.3.2.8 (MPM method) and NEDC-32983P-A, Rev. 2 (GEH method). Peak flux values for individual shell plates 1 and 2 were not listed in the flux evaluations.

d.4 Regulatory Guide 1.190 requires that flux uncertainties be determined to provide confidence in fluence calculations. Total uncertainty values for reactor vessel and reactor vessel internals flux for both pre-EPU (MPM method) and post-EPU (GEH method) operating conditions are included in the respective fluence calculations. Both methods are consistent with Regulatory Guide 1.190 and have been approved by the NRC as described in UFSAR Section 4.3.2.8 (MPM method) and NEDC-32983P-A, Rev. 2 (GEH method). No specific guidance is provided in RG 1.190 regarding the combination of fluence results from multiple methods, but if both pre-EPU and post-EPU fluence values are the results of qualified methods, it is expected that the combination of these values is acceptable with respect to the uncertainty treatment specifications of RG 1.190.

e. In accordance with Paragraph 3 of Section 5.4 of BWRVIP-86, Revision 1, post-EPU fluence calculations were performed for the GGNS vessel using a benchmarked fluence methodology. The methodology was benchmarked using vessel dosimetry from other reactors in accordance with the Boiling Water Reactor Vessel and Internals Program (BWRVIP) Integrated Surveillance Program. In accordance with Paragraph 2 of Section 5.4 of BWRVIP-86, Revision 1, the fluence projection was not based on first cycle dosimetry from GGNS since a major change in core design or management was undertaken with implementation of extended power uprate.

LRA Section 4.2.1 is revised to include discussion of MPM fluence calculation methodology with additions underlined and deletions shown with strikethrough.

4.2.1 <u>Reactor Vessel Fluence</u>

Fluence is calculated based on a time-limited assumption defined by the operating term. Therefore, analyses that evaluate reactor vessel neutron embrittlement based on calculated fluence are time-limited assumption analyses (TLAAs).

Based on operating at EPU power level beginning with Cycle 19, the predicted peak high energy (> 1 MeV) neutron fluence for 54 effective full power years (EFPY) is 4.44E+18 n/cm2 at the vessel inner surface. <u>Post-EPU</u>~~The~~ neutron fluence for the welds and shells of 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. <u>Pre-EPU fluence values were generated using the MPM methodology. MPM methodology is approved by the NRC as described in UFSAR Section 4.3.2.8.</u> ~~This~~ <u>Both pre-EPU (MPM) and post-EPU (GEH)</u> methods adhere~~s~~ to the

guidance provided in RG 1.190 (Ref. 4-9). Results of the fluence evaluation are shown in Table 4.2-1 and used in the evaluations of USE.

LRA Section A.2.1.1 is revised to include discussion of MPM fluence calculation methodology with additions underlined and deletions shown with strikethrough.

A.2.1.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. Therefore, analyses that evaluate reactor vessel neutron embrittlement based on calculated fluence are time-limited aging analyses.

~~The~~ P~~p~~ost-EPU high-energy (> 1 MeV) neutron fluence for the nozzles, welds and shells of 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. Pre-EPU high-energy (> 1 MeV) neutron fluence for the reactor pressure vessel (RPV) beltline region was generated using the MPM methodology. MPM methodology is approved by the NRC as described in UFSAR Section 4.3.2.8. ~~The~~Both pre-EPU (MPM) and post-EPU (GEH) methods adheres to the guidance prescribed in Regulatory Guide (RG) 1.190. ~~as was described in the EPU submittal. (Reference A.3-3).~~

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 4.7.3-1 RESPONSE

a. ASME codes do not have specific requirements for evaluating irradiation effects on RPV internals. The design specification provision to evaluate the fluence effect on core support structure (CSS) components is a General Electric-Hitachi (GEH) provision. CSS components were evaluated for adherence with GE design specification 22A4052, Core Support Structure.

   GEH evaluated the non-CSS components of jet pump beam bolt and shroud head studs. These non-CSS components are further away from the active fuel in the reactor core than the CSS components, hence receive less neutron fluence. Stress relaxation due to fluence effects on the non-CSS bolting components was evaluated and found acceptable.

b. The 40-year and 60-year fluence levels for reactor vessel internals components are calculated in accordance with the NRC-approved fluence methodology of licensing topical report NEDC-32983P-A, Revision 2, and are provided as follows.

Component	40-Year Fluence (n/cm^2)	60-Year Fluence (n/cm^2)
Shroud	1.73E21	2.88E2
Shroud Support Cylinder	1.66E13	2.61E13
Shroud Support Plate	<2.76E13[(1)]	<4.63E13[(1)]
Shroud Support Legs	<5.59E12[(1)]	<8.82E12[(1)]
Core Plate	3.92E20	6.57E20
Top Guide	3.13E21	5.26E21
Core Plate Bolt (Average)	1.72E19	2.71E19
Core Plate Bolt (Peak)	1.42E20	2.23E20
Core Plate Bolt Nut	7.89E19	1.25E20
Core Plate Wedges	8.24E20	1.30E21
Top Guide Bolt (Average)	8.51E19	1.34E20
Top Guide Bolt (Peak)	2.27E20	3.59E20
Top Guide Bolt Nut	5.06E19	7.97E19
Top Guide Bolt Pin	1.14E20	1.79E20
Control Rod Drive Housing	<2.76E13[(1)]	<4.63E13[(1)]
Control Rod Guide Tube	5.04E20	8.46E20
Orificed Fuel Support	5.73E18	2.52E21
Core Spray Sparger	1.50E21	9.04E18
Shroud Head Dome	<1.50E21[(1)]	<9.04E18[(1)]
Shroud Head Stud	<1.50E21[(1)]	<9.04E18[(1)]
Jet Pump Riser Brace	5.93E20	9.36E20
Jet Pump Beam Bolt	5.03E19	7.94E19
Jet Pump Riser	6.15E20	9.70E20
Jet Pump Diffuser	1.05E21	1.66E21

Note:

(1) These components were located at axial positions outside of the region of the EPU flux distribution. For these components, flux was conservatively determined using the flux values for the nearest available node from the EPU flux distribution. Given the conservative nature of this assumption, the actual fluence is definitively less than the listed values.

As identified in specification 22A4052, the shroud was built to the primary stress criteria of ASME Section III subsection NG. ASME codes do not have specific requirements for evaluating

the irradiation effects on RPV internals. The effect of increased fast fluence is an increase in yield and tensile strength. Therefore the fluence effect has no impact on stress qualification of RPV internals. Fast fluence also can cause stress relaxation. Bolting in CSS and non-CSS components was evaluated for stress relaxation and found acceptable.

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.

RAI B.1.11-1 RESPONSE

As stated in LRA Section B.1.11, the Boiling Water Reactor (BWR) Vessel Internals Program is consistent with the program described in NUREG-1801, Section XI.M9, BWR Vessel Internals, without exception. Therefore, by reference, the BWR Vessel Internals Program incorporates the relevant staff-approved BWRVIP documents consistent with NUREG-1801 guidance. The BWRVIP is an industry program providing for implementation of guidelines for inspection and evaluation, repair, water chemistry, and other activities supporting assurance of continued integrity of BWR primary components. As indicated in the July 29, 1997 letter from Brian W. Sheron (NRC) to Carl Terry (BWRVIP Chairman), the U.S. BWR fleet, including GGNS, is committed to comply with BWRVIP guidelines. Thus, the LRA Section A.1.11 reference to "applicable industry standards and staff approved BWRVIP documents" provides a more comprehensive definition of applicable guidance to ensure program effectiveness than to list only specific BWRVIP documents that may be revised or superseded in the future. For example, NUREG-1801, Revision 2 refers to BWRVIP-190 for BWR water chemistry guidelines while NUREG-1801, Revision 1 referred to BWRVIP-29 for BWR water chemistry guidelines.

The LRA Section A.1.11 reference to "applicable industry standards and staff approved BWRVIP documents" is consistent with NUREG-1800, Table 3.0-1, FSAR Supplement for Aging Management of Applicable Systems. NUREG-1800, Table 3.0-1 program descriptions reference specific documents and standards for those programs with a limited set of basis documents. However, for the XI.M9, BWR Vessel Internals program description, Table 3.0-1 refers to "applicable and staff-approved BWRVIP documents" to represent the extensive set of standards and documents that provide the program bases.

To reaffirm the GGNS commitment to the BWRVIP, the following paragraph will be added to LRA Section A.1.11. Additions are marked with underline.

A.1.11 BWR Vessel Internals Program

The BWR Vessel Internals Program manages cracking, loss of material, and reduction of fracture toughness for BWR vessel internal components using inspection and flaw evaluation. This program also provides (1) determination of the susceptibility of cast austenitic stainless steel components, (2) accounting for the synergistic effect of thermal aging and neutron irradiation, and (3) implementation of a supplemental examination program, as necessary. Applicable industry standards and staff-approved BWRVIP documents are used to delineate the program.

The BWRVIP is an industry program providing for implementation of guidelines for inspection and evaluation, repair, water chemistry, and other activities supporting assurance of continued integrity of BWR primary components. As indicated in the July 29, 1997 letter from Brian W. Sheron (NRC) to Carl Terry (BWRVIP Chairman), the U.S. BWR fleet, including GGNS, is committed to comply with BWRVIP guidelines.

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.

<u>Request</u>. 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.1.11-2 RESPONSE

The enhancement stated in LRA Section B.1.11, BWR Vessel Internals, for the management of loss of fracture toughness due to neutron irradiation and thermal aging embrittlement is consistent with the guidance provided in NUREG-1801, Rev. 2, Section XI.M9, BWR Vessel Internals. This guidance establishes the overall parameters of this aspect of the BWR Vessel Internals Program. Details of this aspect of the program remain to be determined, including:

a. the specific scope of components susceptible to neutron irradiation and thermal aging embrittlement,
b. the inspection techniques to be used,
c. sequence of inspections to be conducted, and
d. the methods for evaluating inspection results and extrapolating those results to inaccessible components.

These details will be developed as part of the implementation of the program enhancement described in LRA B.1.11. No revisions to UFSAR Supplement A.1.11 are warranted.

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.1.11-3 RESPONSE

As stated in LRA Section B.1.11, the BWR Vessel Internals Program is consistent with the program described in NUREG-1801, Section XI.M9, BWR Vessel Internals, without exception. Therefore, by reference, the GGNS BWR Vessel Internals Program incorporates the relevant staff-approved BWRVIP documents consistent with NUREG-1801 guidance. The BWRVIP is an industry program providing for implementation of guidelines for inspection and evaluation, repair, water chemistry, and other activities supporting assurance of continued integrity of BWR primary components. As indicated in the July 29, 1997 letter from Brian W. Sheron (NRC) to Carl Terry (BWRVIP Chairman), the U.S. BWR fleet, including GGNS, is committed to comply with BWRVIP guidelines. Consistent with that commitment, the GGNS BWR Vessel Internals Program uses the appropriate toughness versus fluence relationships and flaw evaluation methods from BWRVIP-100-A, for irradiated stainless steel reactor internals where applicable.

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.1.11-4 RESPONSE

1. BWRVIP-139-A includes descriptions of the various steam dryer configurations in use at the time the document was prepared. It provides broad inspection guidelines based on the general characteristics of the various dryer configurations, without establishing bounding criteria that would exclude dryers of similar design. A description of the new steam dryer was included in Attachment 11B to GNRO-2010/00056, License Amendment Request, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1, dated September 8, 2010. Like the original dryer, the new dryer is a curved hood six-bank design. The inspection requirements of BWRVIP-139-A are directly applicable to the GGNS replacement steam dryer.

2. As described in the response to RAI 8, part b, in Attachment 1 of GNRO-2011/00101, Request for Additional Information Regarding Extended Power Uprate Grand Gulf Nuclear Station, Unit 1, dated November 14, 2011, Entergy plans to follow the inspection recommendations of BWRVIP-139 Section 5.3.3 for curved hood steam dryers, and the re-inspection guidelines of Section 5.3.4.

   In GNRO-2012/00031, Supplemental Information – License Conditions, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1, dated April 26, 2012, Entergy proposed license condition 46 which included the following.

   (f) During the first two scheduled refueling outages after reaching full EPU conditions, Entergy shall conduct a visual inspection of all accessible, susceptible locations of the steam dryer in accordance with BWRVIP-139 and GE inspection guidelines. Entergy shall report the results of the visual inspections of the steam dryer to the NRC staff within 60 days following startup.

   (g) At the end of the second refueling outage following the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results for NRC review and approval.

Consistent with NUREG-1801, XI.M9, steam dryer inspection and evaluation will be based on BWRVIP-139-A. Following the second refueling outage after implementation of the extended power uprate, the BWRVIP-139-A recommendations will be supplemented by any additional inspection requirements determined for the long-term steam dryer inspection plan submitted for NRC review and approval in accordance with License Condition 46 part (g). Consequently, GGNS LRA Section B.1.11 need not be modified to include an exception. GGNS LRA Section A.1.11 is modified to note the supplementary steam dryer inspection requirements that may result from License Condition 46 part (g).

LRA Appendix A, Section A.1.11 is revised as shown below. Additions are underlined.

A.1.11 BWR Vessel Internals Program

The BWR Vessel Internals Program manages cracking, loss of material, and reduction of fracture toughness for BWR vessel internal components using inspection and flaw evaluation. This program also provides (1) determination of the susceptibility of cast austenitic stainless steel components, (2) accounting for the synergistic effect of thermal aging and neutron irradiation, and (3) implementation of a supplemental examination program, as necessary. Applicable industry standards and staff-approved BWRVIP documents are used to delineate the program.

Steam dryer inspection and evaluation are based on BWRVIP-139-A. Following the second refueling outage after implementation of the extended power uprate, the BWRVIP-139-A recommendations will be supplemented by any additional inspection requirements determined for the long-term steam dryer inspection plan submitted for NRC review and approval in accordance with License Condition 46 part (g).

The BWR Vessel Internals Program will be enhanced as follows.

- 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.

Enhancements will be implemented prior to the period of extended operation.

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.

RAI B.1.11-5 RESPONSE

The replacement steam dryers at the Susquehanna nuclear site Units 1 and 2 were inspected in 2010 and 2011 respectively. As described in Section 5 of Attachment 11B to GNRO-2010/00056, License Amendment Request, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1, dated September 8, 2010, the March 2010 Unit 1 inspections resulted in changes implemented for the GGNS replacement steam dryer, including revised fabrication procedures and the elimination of tack welds. As described in the response to RAI 12 in Attachment 1 to GNRO-2012/00011, Response to Request for Additional Information Regarding Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1, dated February 20, 2012, the April 2011 Unit 2 inspections resulted in corrective actions to the GGNS dryer manufacturing process.

In GNRO-2012/00031, Supplemental Information – License Conditions, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1, dated April 26, 2012, Entergy proposed license condition 46. License condition 46 delineates requirements for a power ascension testing program, including verifying the continued structural integrity of the steam dryer. License condition 46 also includes the following.

(f) During the first two scheduled refueling outages after reaching full EPU conditions, Entergy shall conduct a visual inspection of all accessible, susceptible locations of the steam dryer in accordance with BWRVIP-139 and GE inspection guidelines. Entergy shall report the results of the visual inspections of the steam dryer to the NRC staff within 60 days following startup.

(g) At the end of the second refueling outage following the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results for NRC review and approval.

As stated in the response to RAI B.1.11-4, steam dryer inspections during the period of extended operation will be conducted in accordance with the BWRVIP Reactor Vessel Internals Program, including the requirements of BWRVIP-139-A and any additional inspection requirements identified by item (g) of license condition 46