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GNRO-2012/00071

July 3, 2012

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

SUBJECT: Response to Request for Additional Information (RAI) dated June 5, 2012
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCE:
1. NRC Letter, "Requests for Additional Information for the Review of the Grand Gulf Nuclear Station, License Renewal Application," dated June 5, 2012 (GNRI-2012/00128) (ML 12137A296)
2. Grand Gulf Nuclear Station, Unit 1 Letter (GNRO-2012/00054), "Response to Request for Additional Information (RAI) dated May 3, 2012"
3. Grand Gulf Nuclear Station, Unit 1 Letter (GNRO-2012/00055), "Response to Request for Additional Information (RAI) Set 12 dated May 9, 2012"

Dear Sir or Madam:

Entergy Operations, Inc is providing, in Attachment 1, the response to the referenced Request for Additional Information (RAI). Attachment 2 includes a revised response to RAI's B.1.27-1, B.1.42-3 and B.1.42-5 provided in a GGNS letter dated May 30, 2012 (Reference 2), a revised response to RAI B.1.24-1 in a GGNS letter dated June 6, 2012 (Reference 3), and editorial corrections to section 3 tables.

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 3rd day of July, 2012.

Sincerely,

MP/jas

Attachments(s): (see next page)

Attachment(s):
1. Response to Request for Additional Information (RAI)
2. Revised Responses to RAI's B.1.27-1, B.1.42-3, B.1.42-5, B.1.24-1 and Editorial Corrections

cc: with Attachment(s)

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Attachment 1 to

GNRO-2012/00071

Response to Request for Additional Information (RAI)

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 4.1-1 RESPONSE

a. The reactor pressure vessel (RPV) subcomponents fabricated to SA-508, Class 2 specifications clad with stainless steel are the closure flange on the vessel and closure flange on the top head. No other reactor pressure vessel subcomponents are fabricated to SA-508, Class 2 specifications with an associated welded cladding.
b. GGNS UFSAR Section 5.3.1.6.4 states, "since reactor vessel specifications require that all low alloy steel pressure vessel boundary material be produced to fine-grain practice, underclad cracking is of no concern." No other reference to underclad cracking is included in the CLB.
c. There is no underclad cracking analysis for GGNS. Since an underclad cracking analysis is not contained or incorporated by reference into the CLB, there is no associated TLAA.

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 4.1-2 RESPONSE

a. The spent fuel cask and new fuel handling cranes were designed to meet applicable criteria of the Crane Manufacturer's Association of America, Inc. (CMAA-70).

   As stated in UFSAR, Section 9D.3.1, Item 3f, the containment hatchway crane load bearing parts were analyzed by GGNS for all applicable loads in accordance with the requirements of the AISC Steel Construction Manual, 7th Edition.

b. The spent fuel cask crane, new fuel handling crane, and polar crane are designed in accordance with CMAA-70. CMAA-70 establishes allowable stress ranges for various materials and configurations for cranes of different service classes. The spent fuel cask crane, new fuel handling crane, and polar crane are designed for a minimum of 100,000 cycles.

   The containment hatchway crane does not have an established limit on the number of times the crane could be used to lift a limiting load.

c. The allowable cycles based on CMAA-70 allowable stress ranges are not time limited and are well above the estimated number of cycles for the spent fuel cask crane, new fuel handling crane, and polar crane during 60 years of plant operation. Therefore, there are no TLAAs associated with crane cycles at Grand Gulf.

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 4.1-3 RESPONSE

a. UFSAR Section 3A states that a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing was developed in compliance with RG 1.20. In addition, UFSAR Section 15D.9 describes Grand Gulf as a fully instrumented prototype BWR/6 251 plant in accordance with RG 1.20, with three-phase testing compliant with RG 1.20 for prototype reactor internals described in UFSAR Section 3.9.2.4. No "sister-plant" flow-induced vibrational assessments are relied upon in the design basis to meet RG 1.20. The term "sister-plant" is not used in the GGNS UFSAR.

b. No "sister-plant" assessments are relied upon for design basis. Therefore, no analyses from other plants are considered TLAA.

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 4.1-4 RESPONSE

a. The plant-specific fracture mechanics evaluation establishes future feedwater nozzle reinspection intervals based on the alternate requirements specified in (GE) NE-523-A71-0594. The plant-specific fracture mechanics evaluation satisfies 10CFR54.3(a)(1, 2, 4, 5, and 6). The plant specific fracture mechanics evaluation does not involve time-limited assumptions defined by the current term of operation, and therefore does not satisfy 10CFR54.3(a)(3).

b. The intent of the plant-specific fracture mechanics evaluation was to justify the inspection intervals in the GE generic evaluation. The specific evaluation concluded that the results presented therein were valid for use in establishing future feedwater nozzle reinspection intervals based on the alternate requirements specified in the generic evaluation. Therefore the specific evaluation did not qualify the feedwater nozzles for a fixed term and is not a 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 4.1-5 RESPONSE

a. An exemption for containment airlock testing was granted for the term of the operating license as described in Supplemental Safety Evaluation Report (SSER) #7, Section 6.2.6. This exemption is due to a requirement in Paragraph III.D.2(b)(ii) of 10 CFR 50 Appendix J that the licensee deemed a hardship.

   The hardship occurs since the existing airlock doors are designed so that a full-pressure test of an airlock can be performed only after strong backs (structural bracing) have been installed on the inner door and would "extend the duration of plant outages by half a day or more several times a year."

   The exemption was granted providing that instead of the full-pressure test required by paragraph III.D.2(b)(ii), the licensee shall, before entering operating modes requiring containment integrity, perform a seal leakage test that satisfies the requirements of paragraph III.D.2(b)(iii), and providing that no maintenance has been performed on the airlock.

   No other exemptions discussed in Clause 2.D of the operating license remain in effect.

b. The justification provided in support of this exemption did not rely on a time-limited analysis or assumption.

c. Since the original justification provided in support of this exemption did not rely on a time-limited analysis or assumption, it need not be identified as a TLAA.

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 4.1-6 RESPONSE

a. Each of the analyses in section 4 considered the post EPU operating conditions and parameters.
    - For LRA Section 4.2, reactor vessel neutron embrittlement evaluations include consideration of increased EPU fluence.
    - For LRA Section 4.3, fatigue analysis results include reevaluations to reflect EPU operating conditions.
    - For LRA Section 4.4, environmental qualification evaluations include EPU conditions.
    - For LRA Section 4.6, the containment liner plate, metal containments and containment penetrations include considerations of EPU conditions.
    - For LRA section 4.7, the erosion of the MSL flow restrictors, determination of intermediate high energy line break locations, and fluence effects on the reactor vessel internals each include considerations of EPU operating conditions.
b. Justifications are as follows.
    - For LRA Section 4.2, the embrittlement of metals increases with increased fluence, so the consideration of higher total fluence in the TLAA evaluations is bounding.
    - For LRA Section 4.3, EPU does not decrease cyclic stresses. Slightly higher

temperature differences will be experienced under EPU conditions increasing thermal stresses and consequently results will be conservative for fatigue.

- For LRA Section 4.4, EQ components are refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the appropriate evaluation as required by 10 CFR 50.49. Therefore, non-approval of EPU will not adversely affect TLAAs for EQ components.
- For LRA Section 4.6, the analyzed containment temperature and pressure transient was reviewed and found to adequately account for the increase of mass and energy resulting from EPU. Therefore, non-approval of EPU will not affect the containment temperature and pressure transient and, therefore, will not affect the TLAAs that are based on that transient.
- For LRA section 4.7, fatigue evaluations considered the higher flow rates and velocities that would be present under EPU which resulted in slightly higher calculated fatigue usage for some locations. Therefore, existing operating conditions will remain bounded by EPU conditions if EPU is not implemented.

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).

Issue: 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.

Request:

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 4.2.6-1 RESPONSE

a.1 The fracture toughness values are temperature dependent for any given material. As shown in LRA Section 4.2.6, the plant-specific temperature at 1/4T depth into the vessel wall was determined to be 400 degrees fahrenheit (°F) at 300 seconds into the thermal shock event. It was also stated that using the highest adjusted reference temperature (ART), the beltline material reaches upper shelf (200 ksi-in½) at ~158°F. Since this temperature is significantly lower than 400°F, it is assured that the beltline material remains at upper shelf throughout the thermal shock event. Figure 5 of the Ranganath paper, 'Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident,' Ranganath, S., Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, further demonstrates that at 300 seconds into the thermal shock event and 1/4T depth into the vessel wall, the maximum applied stress intensity is 103 ksi-in½. Therefore, there is sufficient margin to prevent fracture due to reflood thermal shock.

a.2 The 53 degree value provided in LRA Section 4.2.6 is for the limiting material considering all materials evaluated in LRA Table 4.2.2. This includes all materials that are exposed to fast fluence values in excess of 1e17 n/cm2.

a.3 A bounding evaluation was performed in the Ranganath paper. The GGNS parameters such as vessel size, vessel thickness, and fluence are similar to or bounded by the input parameters considered in the Ranganath evaluation.

b. There is no reflood thermal shock analysis of the RPV core shroud. The reflood thermal shock analysis addressed in LRA Section 4.2.6 and subsequent Ranganath evaluation apply to the reactor vessel. Effects on the SCC indication have not been evaluated for a reflood thermal shock event since no reflood thermal shock analysis exists for GGNS.

c. The indication found in the shroud circumferential weld was evaluated following discovery in 2005. The evaluation determined an acceptable inspection frequency and did not qualify the affected weld for service to the end of the 40-year operating term. The flaw was found to be acceptable for service until the next inspection even though the evaluation conservatively considered it to be through-wall (or 2.0 inches deep rather than the measured 0.28 inches deep). Therefore, the crack growth analysis

does not meet the 10 CFR 54.3 definition of a TLAA. Potential SCC crack growth in the core shroud is monitored by the ISI Program described in LRA B.1.23.

RAI 4.7.1-1

Background: 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 x 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 x 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.

RAI 4.7.1-1 RESPONSE

The main steam line (MSL) flow restrictor erosion rate of 0.004 inches per year stated in UFSAR Section 5.4.4.4 is highly conservative. Information from a later evaluation bound by proprietary agreement shows the expected flow restrictor erosion to be less than the 0.160 inches derived from the UFSAR rate of 0.004 inches per year for 40 years of operation.
Reduction in the erosion-corrosion rate from the UFSAR value of 0.004 inches per year is attributed to the following.

1. The use of grade CF8 stainless steel with a chromium content of 18 percent (%) to 21% provides resistance to erosion-corrosion damage for the upstream casting in the MSL flow restrictor. Typically, a chromium content of 12% is sufficient to create a passive chromium oxide layer that prevents corrosion of stainless steels in water/steam environments. Therefore, the 18 percent minimum chromium in Grade CF8 is more than sufficient to generate the passive chromium oxide layer for corrosion protection.
2. The relative erosion-corrosion rate for steels begins to reduce at temperatures above approximately 302 °F. GGNS operates with a main steam temperature of greater than 500 °F.
3. Chloride intrusion is not expected to occur in the MSL therefore, the Grade CF8 material will not experience the potential for pitting and/or stress corrosion cracking associated with chloride ions.

Projected erosion after 60 years using the erosion rate from the later evaluation is less than the total erosion after 40 years calculated using the rate stated in the UFSAR. Therefore, this analysis has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Attachment 2 to

GNRO-2012/00071

Revised Responses to RAI's B.1.27-1, B.1.42-3, B.1.42-5, B.1.24-1 and Editorial Corrections

RAI B.1.27-1 Revised Response

In the response to RAI B.1.27-1 provided in letter GNRO-2012/00054 dated May 30, 2012 a change was made to Appendix B.1.27 to revise an enhancement. It has been determined that a corresponding change to Appendix A.1.27 "Masonry Wall Program" is also required. The change to A.1.27 is provided below. Deletions are shown with strikethrough.

The Masonry Wall Program will be enhanced as follows.

- Monitor gaps between the supports and masonry walls that could potentially affect wall qualification.

- Require masonry walls to be inspected every five years~~ unless technical justification is provided to extend the inspection to a period not to exceed ten years~~.

RAI B.1.42-3 Revised Response

In the response to RAI B.1.42-3 provided in letter GNRO-2012/00054 dated May 30, 2012, a change was made to Appendix B.1.42 to add enhancements. It has been determined that a corresponding change to Appendix A.1.42 "Structures Monitoring Program" is also required. The change to A.1.42 to add the following enhancements is provided below. Additions are shown with underline.

- <u>Require direct visual examinations when access is sufficient for the eye to be within 24-inches of the surface to be examined and at an angle of not less than 30° to the surface. Mirrors may be used to improve the angle of vision and accessibility in constricted areas.</u>

- <u>Specify that remote visual examination may be substituted for direct examination. For all remote visual examinations, optical aids such as telescopes, borescopes, fiber optics, cameras, or other suitable instruments may be used provided such systems have a resolution capability at least equivalent to that attainable by direct visual examination.</u>

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

RAI B.1.42-5 Revised Response

In the response to RAI B.1.42-5 provided in letter GNRO-2012/00054 dated May 30, 2012, a change was made to Appendix B.1.42 to add an enhancement. It has been determined that a corresponding change to Appendix A.1.42 "Structures Monitoring Program" is also required. The change to A.1.42 to add the following enhancement is provided below. Additions are shown with underline.

- <u>Include periodically inspecting the leak chase system associated with the upper containment pool and spent fuel pool to ensure the tell-tales are free of significant</u>

blockage. The inspection will also inspect concrete surfaces for degradation where leakage has been observed, in accordance with this Program.

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

RAI B.1.24-1 Revised Response

In the response to RAI B.1.24-1 provided in letter GNRO-2012/00055 dated May 30, 2012, a change was made to Appendix B.1.24 to add to an enhancement on the detection of aging. It has been determined that a corresponding change to Appendix A.1.24 "Inservice Inspection – IWF Program" is also required. The change to A.1.24 to add the following enhancement is provided below. Additions are shown with underline.

- Clarify that detection of aging will include:

  a) Monitoring structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts) and anchor bolts will be monitored for loss of material, loose or missing nuts, loss of preload and cracking of concrete around the anchor bolts.

  b) Volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 should be performed for high strength structural bolting to detect cracking in addition to the VT-3 examination. This volumetric examination may be waived with adequate plant-specific justification.

  c) Identification of component supports that contain high strength bolting (actual measured yield greater than or equal to 150 ksi) in sizes greater than 1 inch nominal diameter. The extent of examination for support types that contain high-strength bolting will be as specified in ASME Code Section XI, Table IWF-2500-1. GGNS will examine high strength structural bolting on the frequency specified in ASME Code Section XI, Table IWF-2500-1.

- Include the following as unacceptable conditions.
  a) Loss of material due to corrosion or wear, which reduces the load bearing capacity of the component support.
  b) Debris, dirt, or excessive wear that could prevent or restrict sliding of the sliding surfaces as intended in the design basis of the support.
  c) Cracked or sheared bolts, including high strength bolts, and anchors.

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

Editorial Corrections

A review of the license renewal application discovered the following edtiorial errors in the application. Additions are shown with underline and strikethroughs are used for deletions.

1. In Table 3.1.2-4-2 Line items for accumulator on page 3.1-78 of the LRA there are two line items, one with NUREG-1801 Item of 3.1.1-40 and one with 3.2.1-40. The 3.1.1-40 line is in error and should not be in the LRA.

Table 3.1.2-4-2
Reactor Recirculation System
Nonsafety-Related Components Affecting Safety-Related Systems
Summary of Aging Management Evaluation

Table 3.1.2-4-2: Reactor Recirculation System Nonsafety-Related Components Affecting Safety-Related Systems								
Component Type	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801 Item**	**Table 1 Item**	**Notes**
~~Accumulator~~	~~Pressure boundary~~	~~Carbon steel~~	~~Air – indoor (ext)~~	~~Loss of material~~	~~External Surfaces Monitoring~~	~~V.E.E-44~~	~~3.1.1-40~~	~~C~~
Accumulator	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	V.E.E-44	3.2.1-40	C
Accumulator	Pressure boundary	Carbon steel	Lube oil (int)	Loss of material	Oil Analysis	V.D2.EP-77	3.2.1-49	C, 104
Bolting	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	Bolting Integrity	IV.C1.RP-42	3.1.1-63	A
Bolting	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of preload	Bolting Integrity	IV.C1.RP-43	3.1.1-67	A
Bolting	Pressure boundary	Stainless steel	Air – indoor (ext)	Loss of preload	Bolting Integrity	IV.C1.RP-43	3.1.1-67	A
Filter housing	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	V.E.E-44	3.2.1-40	C

2. In Table 3.3.2-19-28 on page 3.3-305 of the LRA there is a missing line item of a piping component line with raw water.

Table 3.3.2-19-28: HPCS Diesel Generator System [10 CFR 54.4(a)(2)]

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Flexible connection	Pressure boundary	Stainless steel	Treated water (int)	Loss of material	Water Chemistry Control – Closed Treated Water Systems	VII.C2.A-52	3.3.1-49	C
Flexible connection	Pressure boundary	Stainless steel	Waste water (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	VII.E5.AP-278	3.3.1-95	C
Piping	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.I.A-77	3.3.1-78	A
Piping	Pressure boundary	Carbon steel	Condensation (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	VII.E5.AP-280	3.3.1-95	C
Piping	Pressure boundary	Carbon steel	Fuel oil (int)	Loss of material	Diesel Fuel Monitoring	VII.H1.AP-105	3.3.1-70	A, 303
Piping	Pressure boundary	Carbon steel	Raw water (int)	Loss of material	Service Water Integrity	VII.C1.AP-194	3.3.1-37	C, 304
Piping	Pressure boundary	Carbon steel	Treated water (int)	Loss of material	Water Chemistry Control – Closed Treated Water Systems	VII.H2.AP-202	3.3.1-45	A

3. In Table 3.5.2-1 on page 3.5-66 of LRA there is a containment personnel lock item where the NUREG-1801 line says II.B4.C-16ct. The ct is in error and should not be included.

Table 3.5.2-1: Containment Building

Structure and/or Component or Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Containment cylinder wall electrical penetrations	PB, SSR	Carbon steel	Air – indoor uncontrolled	Loss of material	CII – IWE Containment Leak Rate	II.B4.CP-36	3.5.1-35	A
Containment equipment hatch	EN, MB, PB, SSR	Carbon steel	Air – indoor uncontrolled	Loss of material	CII – IWE Containment Leak Rate	II.B4.C-16	3.5.1-28	B
Containment hatchway crane-crane rails	SNS	Carbon steel	Air – indoor uncontrolled	Loss of material	Inspection of OVHLL	VII B.A-07	3.3.1-52	A
Containment hatchway crane structural girders	SNS	Carbon steel	Air – indoor uncontrolled	Loss of material	Inspection of OVHLL	VII B.A-07	3.3.1-52	A
Containment personnel lock	EN, MB, PB, SSR	Carbon steel	Air – indoor uncontrolled	Loss of material	CII – IWE Containment Leak Rate	II.B4.C-16~~ct~~	3.5.1-28	B
Containment sumps liner and penetrations	EN, PB, SSR	Stainless steel	Exposed to fluid environment	Loss of material	Structures Monitoring	III.A7.T-23	3.5.1-52	E

4. In Table 3.3.2-19-34 there are eleven line items requiring a correction for the NUREG-1801 item from F1 to F2.

Table 3.3.2-19-34: Control Building HVAC System [10 CFR 54.4(a)(2)]

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Blower housing	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.F2~~1~~.A-10	3.3.1-78	A
Blower housing	Pressure boundary	Carbon steel	Air – indoor (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	V.B.E-25	3.2.1-44	C
Bolting	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	Bolting Integrity	VII.I.AP-125	3.3.1-12	A
Bolting	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of preload	Bolting Integrity	VII.I.AP-124	3.3.1-15	A
Damper housing	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.F2~~1~~.A-10	3.3.1-78	A
Damper housing	Pressure boundary	Carbon steel	Air – indoor (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	V.B.E-25	3.2.1-44	C
Duct	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.F2~~1~~.A-10	3.3.1-78	A
Duct	Pressure boundary	Carbon steel	Air – indoor (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	V.B.E-25	3.2.1-44	C
Duct flexible connection	Pressure boundary	Elastomer	Air – indoor (ext)	Change in material properties	External Surfaces Monitoring	VII.F2~~1~~.AP-102	3.3.1-76	A
Duct flexible connection	Pressure boundary	Elastomer	Air – indoor (ext)	Cracking	External Surfaces Monitoring	VII.F2~~1~~.AP-102	3.3.1-76	A

Table 3.3.2-19-34: Control Building HVAC System [10 CFR 54.4(a)(2)]

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Duct flexible connection	Pressure boundary	Elastomer	Air – indoor (int)	Change in material properties	External Surfaces Monitoring	VII.F2~~1~~.AP-102	3.3.1-76	A
Duct flexible connection	Pressure boundary	Elastomer	Air – indoor (int)	Cracking	External Surfaces Monitoring	VII.F2~~1~~.AP-102	3.3.1-76	A
Fan housing	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.F2~~1~~.A-10	3.3.1-78	A
Fan housing	Pressure boundary	Carbon steel	Air – indoor (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	V.B.E-25	3.2.1-44	C
Filter housing	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.I.A-77	3.3.1-78	A
Filter housing	Pressure boundary	Carbon steel	Air – indoor (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	V.D2.E-29	3.2.1-44	C
Heat exchanger (shell)	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.F2~~1~~.AP-41	3.3.1-80	A
Heat exchanger (shell)	Pressure boundary	Carbon steel	Condensation (int)	Loss of material	Internal Surfaces in Miscellaneous Piping and Ducting Components	VII.F2~~1~~.A-08	3.3.1-90	A
Humidifier housing	Pressure boundary	Carbon steel	Air – indoor (ext)	Loss of material	External Surfaces Monitoring	VII.F2~~1~~.A-10	3.3.1-78	A