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GNRO-2012/00103

September 4, 2012

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

SUBJECT: Response to Requests for Additional Information (RAI) Set 31 dated August 7, 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 August 7, 2012 (GNRI-2012/00173)(ML12208A267)
2. Grand Gulf Nuclear, Unit 1 Letter (GNRO-2012/100085), "Response to Request for Additional Information (RAI) Set 25 dated June 27, 2012", dated July 25, 2012

Dear Sir or Madam:

Entergy Operations, Inc is providing, in Attachment 1, the response to the referenced Requests for Additional Information (RAI). Attachment 2 includes a change to the response to RAI 4.7.3-1 part b table provided in reference 2 to correct an error in the table.

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 4th day of September, 2012.

Sincerely,

MP/jas

Attachment(s): (see next page)

Attachment(s): 1. Response to Requests for Additional Information (RAI)
2. Revised response to RAI 4.7.3-1

cc: with Attachments

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Attachment 1 to

GNRO-2012/00103

Response to Requests for Additional Information (RAI)

The format for the RAI responses below is as follows. The Request for Additional Information (RAI) is listed in its entirety as received from the Nuclear Regulatory Commission (NRC) with background, issue and request subparts. This is followed by the Grand Gulf Nuclear Station (GGNS) RAI response to the individual question.

RAI 4.3-5a

Background. In the response to request for additional information (RAI) 4.3-5 dated June 21, 2012, the applicant stated that the design specifications of expansion joints identified a conservative number of cycles at a given amount of expansion so that simplified analyses could be performed. The applicant also stated that the typical analysis shows the expansion joint is qualified for many more cycles than was specified with these simplified bounding assumptions and Entergy Nuclear, Inc. (Entergy) completed a review of these analyses to verify the expansion joints are adequate for 60 years. The applicant also explained seismically qualified expansion joints have cycle specifications for seismic events and expansion joints inside primary containment include cycles due safety relief valve (SRV) lifts.

Issue. The staff noted the concerns identified in RAI 4.3-5 were related to license renewal application (LRA) Section 4.3.2.2, which discusses metal fatigue of non-Class 1 non-piping components. The applicant did not provide the number of cycles that the design specifications identified for non-Class 1 expansion joints. The applicant also did not provide the number of cycles for earthquakes and SRV actuations that are assumed in the design of its non-Class 1 expansion joints. Therefore, the staff cannot verify the adequacy of the disposition of the time-limited aging analysis (TLAA) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1)(i).

Request.

a. Identify all the transients and associated number of cycles that were considered as an input to the fatigue analyses of the non-Class 1 expansion joints discussed in LRA Section 4.3.2.2. In addition, identify all the non-Class 1 expansion joints that are inside primary containment.
b. Describe the details of the review that was completed for these expansion joint fatigue analyses in order to verify the components are adequate for 60-years of operation. Justify that this review and evaluation demonstrates that the number of cycles originally analyzed as part of the expansion joint design will not be exceeded after 60-years of operation, in accordance with 10 CFR 54.21(c)(1)(i).
c. Justify the statement, in response to RAI 4.3-5, that the allowable numbers of cycles for the other transients (i.e., transients used in fatigue analyses other than SRV and seismic) are well beyond reasonably postulated numbers of the transients, making it unnecessary to monitor these transients.
d. Justify the statement, in response to RAI 4.3-5, that the typical analysis shows the expansion joint is qualified for many more cycles than was specified with the simplified bounding assumptions as it relates to transients used in the fatigue analyses, except for SRV and seismic transients.

RAI 4.3-5a RESPONSE

a. The transients involved for expansion joints are the cycles of movements of the expansion joint during normal and upset transients and from earthquakes. The cycles specified for expansion joints varied by the location and use of the expansion joint as well as the level of complexity of the design specification. The following table identifies the cycles that were analyzed and identifies the limiting numbers of cycles for the expansion joints. Earthquake cycles have not occurred and are tracked separately; consequently they are not included in the table. The expansion joints within the primary containment are in the standby liquid control system, the compressed air system, and at the emergency core cooling system (ECCS) suction strainer.

System (location description)	**Design Cycles**	**Evaluation**
Standby liquid control (at tank outlet and pump inlets)	<1000 cycles of 0.125 inch movement	Qualified for over 400,000 cycles of movement (0.125 inch).
Standby liquid control (at test tank)	600 thermal cycles 600 pressure cycles	This nonsafety-related expansion joint for the test tank is only used for testing and it therefore experiences no thermal or pressure transients during normal operation. The expansion joint was shown to be qualified for over 1700 cycles.
Standby liquid control (at pump discharge)	1500 pressure cycles 1500 thermal cycles 13,000 SRV actuations Infinite pulsations	Usage factor for these cycles is 0.1, so the expansion joints are acceptable for many more cycles than specified.
ECCS suction strainers (in suppression pool)	12,600 cycles combined loadings from, seismic, SRV operation and LOCA as well as unlimited chugging cycles	As shown in LRA Table 4.3-1, the projected number of cycles from SRV operation and earthquakes is significantly less than 12,600 cycles.
Standby diesel generators (turbocharger water outlets)	1500 thermal cycles, infinite number of cycles for vibration loading	Usage factor for these cycles is 0.04, so expansion joints are acceptable for many more cycles than specified.
High pressure core spray (standby service water supply to diesel)	750 thermal cycles continuous vibration	Usage factor for these cycles was 0.043, so expansion joints are acceptable for many more cycles than specified.
High pressure core spray (diesel air start)	750 thermal cycles continuous vibration	Usage factor for these cycles was 0.0265, so expansion joints are acceptable for many more cycles than specified.
High pressure core spray (diesel fuel oil)	750 thermal cycles	Usage factor for these cycles was 0.011, so expansion joints are acceptable for many more cycles than specified.
High pressure core spray (non-safety related diesel air intake)	750 thermal cycles vibration	This portion of the air intake is not heated during diesel operation. These expansion joints at the air intake are not expected to experience thermal cycles; therefore the nominal value specified is adequate.

System (location description)	Design Cycles	Evaluation
High pressure core spray (non-safety related diesel cooling water drain)	750 thermal cycles vibration	This isolated portion of the drain line is not heated during diesel operation. These expansion joints at the drain are not expected to experience thermal cycles and therefore the nominal value specified is adequate.
High pressure core spray (diesel exhaust)	1200 thermal cycles	The joints were determined to be qualified for over 4000 cycles; so expansion joints are acceptable for many more cycles than specified.
Leakage detection and control system (exhaust blower inlet)	500 thermal cycles 500 dynamic cycles	Usage factor for these cycles was 0.0051, so expansion joints are acceptable for many more cycles than specified.
Compressed air (air accumulators)	400 thermal 300 dynamic	Over 10,000 cycles were determined to be allowed, so expansion joints are acceptable for many more cycles than specified.
Heater, vents, and drains System /NSR affecting SR (a2) (at heater drain pumps)	600 thermal cycles	This system will be heated up as part of plant startup. As shown in LRA Table 4.3-1, plant startups are limited to less than 120. Therefore, cycles will be much less than specified.
Condensate and feedwater system /NSR affecting SR (a2) (at condensate pumps)	600 thermal cycles	This system will be heated up as part of plant startup. As shown in LRA Table 4.3-1, plant startups are limited to less than 120. Therefore, cycles will be much less than specified.
Extraction steam system / NSR affecting SR (a2) (turbine extraction steam)	2000 cycles	This system will be heated up as part of plant startup. As shown in LRA Table 4.3-1, plant startups are limited to less than 120. Therefore, cycles will be much less than specified.
Condensate and feedwater system /NSR affecting SR (a2) (feedpump turbine exhaust)	3000 cycles	This system will be heated up as part of plant startup. As shown in LRA Table 4.3-1, plant startups are limited to less than 120. Therefore, cycles will be much less than specified.

b. The review is summarized in the evaluation column in response part “a”.

c. The assumed numbers of design cycles for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) remain adequate for the period of extended operation (PEO) since no OBE or SSE earthquakes have occurred. Details provided in part “a” response provide basis for concluding that allowable numbers of transient cycles for expansion joints are well beyond reasonably postulated numbers of transients for operation through the PEO, such that monitoring these transients is unnecessary.

d. Refer to evaluation column of part “a” response.

RAI 4.3-9a

Background. In the response to RAI 4.3-9 dated June 21, 2012, revised LRA Section A.2.2.3 states that industry-accepted techniques will be used for consideration of the effects on fatigue of the reactor water environment (environmentally assisted fatigue - EAF), including techniques for incorporating the impact of dissolved oxygen concentration into the calculation of fatigue environmental correction factors.

Issue. The staff noted that it is possible these industry-accepted techniques may not or will not be approved for use by the staff. Furthermore, the staff noted that the concern identified in RAI 4.3-9 specifically addressed the use of a time-weight percentage for hydrogen water chemistry/normal water chemistry (HWC/NWC) in the formulation of F_{en} values would underestimate the environmentally-adjusted cumulative usage factors (CUFs). This underestimate in the CUF_{en} may potentially be non-conservative for carbon steel/low alloy steel components. The applicant has not explained how the use of unspecified industry-accepted techniques will resolve this potential non-conservatism.

Request.

a. Revise LRA Section A.2.2.3 to indicate that future calculation of F_{en} values will incorporate available transient cycle occurrence data during operating times in NWC and HWC instead of using a time-weight percentage for NWC and HWC operation.

b. In lieu of revising LRA Section A.2.2.3, justify the appropriateness of relying upon the use of industry-accepted techniques that are not clearly specified and have not been approved by the staff for incorporating the impact of dissolved oxygen concentration into the calculation of fatigue environmental correction factors.

RAI 4.3-9a RESPONSE

a. LRA Section A.2.2.3 is revised as shown below to indicate that future calculation of F_{en} values will incorporate available transient cycle occurrence data during operating times in NWC and HWC instead of using a time-weighted percentage for NWC and HWC operation.

b. Since the LRA is revised, no additional response required for part b.

Change the next to last paragraph of LRA section A.2.2.3. Additions are shown with underline.

A.2.2.3 Effects of Reactor Water Environment on Fatigue Life

Future calculation of F_{en} values will incorporate available transient cycle occurrence data during operating times in normal water chemistry (NWC) and hydrogen water chemistry (HWC). If an acceptable CUF cannot be calculated, GGNS will repair or replace the affected locations before exceeding an environmentally adjusted CUF of 1.0.

RAI B.1.34-2a

Background. LRA Section B.1.34 states that the applicant has not experienced cracking in its ASME Code Class 1 small bore piping. However, no specific information was provided regarding the search and the source of its operating experience. The staff issued RAI B.1.34-2 in a letter dated April 18, 2012. The RAI requested that the applicant to identify the specific sources of information reviewed and describe the process or methodology used to find potential instances of cracking in Class 1 small-bore piping.

The applicant provided its response to RAI B.1.34-2 in a letter dated May 18, 2012. In its response, the applicant stated that the specific sources of information reviewed were the paperless condition reporting system (PCRS), the NRC Licensee Event Reports (LER) database submitted from Entergy nuclear facilities and interviews with its plant staff. It further stated that its PCRS search, "included a review of ten years of operating experience contained in the PCRS."

Issue. The staff noted that although the plant has been in operation for approximately 27 years, the applicant's search of its plant-specific operating experience using its database, PCRS, was limited to 10 years. In addition, supplementing the PCRS database review with information obtained from the NRC LER database the information may not be sufficient or complete, since the NRC LER database, which only includes events that meet reporting criteria outlined in 10 CFR 50.72 and 50.73, may not be all inclusive. The staff also determined that interviews of plant personnel, while potentially providing additional context for previously documented information, does not constitute an established repository of operating experience information.

Request. The staff requests that the applicant provide a technical basis for limiting the review of plant-specific operating experience contained in the PCRS, or other plant databases, to the previous 10 years of operation for the One-Time Inspection - Small-Bore Piping Program.

RAI B.1.34-2a RESPONSE

Since this program is applicable to class 1 piping components, relevant operating experience would be associated with leaks of class 1 reactor coolant pressure boundary piping components. Reactor coolant pressure boundary leakage requires submittal of a Licensee Event Report (LER) in accordance with 10 CFR 50.73. As a result, the LER search provides a complete source of operating experience for identification of through-wall cracking of ASME Code Class 1 small bore piping components.

The review of PCRS information for greater than a ten-year period would provide no additional information on leakage from ASME Code Class 1 small bore piping than that obtained through the LER search. Nevertheless, the technical basis for a ten-year review of plant-specific operating experience is provided in Section 4.4 of NEI 95-10 Revision 6 "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," which reads as follows.

> *Operating Experience - Aging Effects Requiring Management*
>
> *A plant-specific operating experience review should assess the operating and maintenance history. A review of the prior five to ten years of operating and maintenance history should be sufficient.*

In Regulatory Guide 1.188 Revision 1, the NRC staff found NEI 95-10 Revision 6 acceptable for use in implementing the license renewal rule, without exceptions. The basis for this NEI guidance stems from the fundamental nature of the effects of aging. Since degradation due to aging mechanisms is a progressive phenomenon, the rate of failures due to aging would be expected to increase with time. A failure that may have occurred greater than ten years ago, yet has not recurred since that time, is unlikely attributable to age-related degradation.

Attachment 2 to

GNRO-2012/00103

Revised Response to RAI 4.7.3-1

In the response to RAI 4.7.3-1 part b, a table was provided for the 40 and 60 year fluence values of the reactor vessel internals. An error was found in the first line of the table for the 60 year fluence value of the shroud. The table provided in the response listed this fluence value as 2.88E2 when it should have been 2.88E21. A corrected table is provided below with the change underlined as an addition.

Component	40-Year Fluence (n/cm^2)	60-Year Fluence (n/cm^2)
Shroud	1.73E21	2.88E2<u>1</u>
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.