Entergy

Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

Kevin J. Mulligan
Vice President, Operations
Grand Gulf Nuclear Station
Tel. (601) 437-7500

Enclosure 1 Contains PROPRIETARY Information

GNRO-2013/00003

January 18, 2013

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

SUBJECT: Response to Requests for Additional Information (RAI) 4.2.1-2b and 4.7.3-1a
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCE: Entergy Operations Letter, "Response to Requests for Additional Information (RAI) Set 42, dated December 18, 2012 (GNRO-2012/00156)

Dear Sir or Madam:

Entergy Operations, Inc. is providing in the attachment, the response to the Requests for Additional Information (RAI) 4.2.1-2b and 4.7.3-1a as discussed in the referenced letter.

This letter contains no new commitments.

The response to RAI 4.7.3-1a contains proprietary information which is located in Enclosure 1. Enclosure 1 is requested to be withheld from public disclosure in accordance with 10CFR 9.17(a)(4) and 10CFR2.390(a)(4) and is PROPRIETARY to General Electric Hitachi Nuclear Energy (GEH). An affidavit attesting to the proprietary nature of the information is provided in Enclosure 3. A non-proprietary version of Enclosure 1 is included in Enclosure 2.

If you have any questions or require additional information, please contact Jeffery A. Seiter at 601-437-2344.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of January, 2013.

Sincerely,

KJM/jas

Attachment and Enclosures: (see next page)

Attachment: Response to Requests for Additional Information (RAI)

Enclosures 1 1-2OR45T-13 GEH Responses to Grand Gulf License Renewal RAIs GEH Proprietary Information-Class III (Confidential)

2 1-2OR45T-13 GEH Responses to Grand Gulf License Renewal RAIs GEH Non-proprietary - Class I (Public)

3 1-2OR45T-13 Affidavit

cc: with Attachment and Enclosures

Mr. John P. Boska, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, DC 20555

cc: without Attachment and Enclosures

Mr. Elmo E. Collins, Jr.
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, TX 76011-4511

U.S. Nuclear Regulatory Commission
ATTN: Mr. Alan Wang, NRR/DORL
Mail Stop OWFN/8 B1
11555 Rockville Pike
Rockville, MD 20852-2378

U.S. Nuclear Regulatory Commission
ATTN: Mr. Nathaniel Ferrer NRR/DLR
Mail Stop OWFN/ 11 F1
11555 Rockville Pike
Rockville, MD 20852-2378

NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Attachment to

GNRO-2013/00003

Response to Requests for Additional Information (RAI)

The format for the Requests for Additional Information (RAI) responses below is as follows. The 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 4.2.1-2b

Background. In its License Renewal Application, the applicant used a reactor vessel fluence estimate that was obtained by combining a pre-extended power uprate (EPU) fluence value obtained from MPM Technologies calculations to a post-EPU fluence value obtained from General Electric Hitachi (GEH).

By letter dated October 15, 2012, the applicant responded to RAI 4.2.1-2a that, in part, addressed the discrepancies between the pre-EPU and post-EPU flux values for the core shroud welds (H1, V1, V2, V3, and V4 welds in the top portion of the core shroud). In the applicant's analysis for these weld locations, the pre-EPU peak flux values are greater than the post-EPU peak flux values approximately by three orders of magnitude (i.e., approximately 1.0E10 n/cm^2-s versus 1.0E07 n/cm^2-s). The core shroud neutron flux values are described in the applicant's response, dated July 25, 2012, to RAI 4.2.1-2.

In addition, RAI 4.2.1-2a addressed the issue regarding the combined analytic uncertainty associated with the two different methods for the pre-EPU and post-EPU fluence calculations (i.e., MPM method and GEH method, respectively).

Issue. The applicant's response to RAI 4.3.1-2a states that the MPM approach used a bounding-case flux at the core shroud weld locations in the top portion of the core shroud shell. However, the applicant's response does not provide specific information to justify why the flux values obtained from the MPM method are bounding-case flux values for the core shroud weld locations.

In addition, the applicant's response indicates the following with respect to the combined analytic uncertainty:

- The applicant acknowledged that proving the independence of all uncertainty terms in both fluence calculations is not possible.
- The applicant stated that because the independence of the uncertainty values cannot be proven, relative uncertainties are considered instead.
- The applicant expressed total uncertainty as a "relative uncertainty" combination, which is, in actuality, a weighted average of the uncertainties associated with both methods.

The staff noted the following issues with the applicant's response:

- The applicant's approach is not a valid or acceptable way to perform an analytic uncertainty analysis because it is not consistent with Regulatory Guide (RG) 1.190.
- The errors associated with both fluence calculation methods will propagate when the two fluence calculational methods are combined.
- The uncertainty associated with adding the two fluence values together should be higher than either as a stand-alone calculation.

Request.

a. Provide additional information to clarify why the flux values obtained from the MPM method are bounding-case flux values for the core shroud weld locations (i.e., H1, V1, V2, V3, and V4 welds). As part of the response, clarify which locations are actually associated with the bounding-case flux values and why the bounding-case flux values have sufficient conservatisms as bounding-case values.

b. Provide an analytic uncertainty analysis for the combined fluence values, consistent with RG 1.190 that states that each of the fluence methods be supported by an analytic uncertainty analysis. Alternatively, provide reactor vessel neutron fluence values that were calculated using a single, NRC-approved method that has a valid and accepted analytic uncertainty analysis, and reconfirm the validity of the related neutron embrittlement time-limited aging analysis (TLAAs) in light of the updated fluence values.

RAI 4.2.1-2b RESPONSE:

a. The flux values generated by the GEH method for core shroud welds H1, V1, V2, V3, and V4, were derived using the DORT code reviewed and approved by the NRC in NEDC-32983P-A, Revision 2, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," January 2006. The GEH method produced flux values at the top guide inside radius that continually decline with increasing height above the active fuel.

The method used by MPM Technologies, Inc. (the MPM method) to generate flux values for core shroud welds H1, V1, V2, V3, and V4 is based on a modified synthesis procedure to account for the geometry and shielding in the area above the active fuel. The MPM calculated flux at H1 is approximately 50% higher than the MPM value at weld H2. The fact that flux at weld H1is higher than the flux at H2, a location approximately 34" closer to the active fuel, is an indication that the values at the higher elevations of the top guide are conservative and provide a bounding-case solution.

The variation of modeling between the two methods resulted in significantly higher flux values for welds H1, V1, V2, V3, and V4 using the MPM method. Possible contributing factors to the differing results include void fraction assumptions above the active fuel, upper guide structure modeling, and water column height in the area between the fuel and shroud. As the staff comments on BWRVIP-145-A indicate, there is significant difficulty in using the conventional discrete ordinates method in calculating fluence values for components away from the middle plane of the core.

Shroud welds H1, V1, V2, V3, and V4 are the only welds in the vessel or shroud where pre-EPU flux values were higher than post-EPU flux values. The most important top guide locations for considering the effects of fluence are areas at the lower elevations of the top guide where the highest flux and fluence occur near the center of the core. Specifically, the BWRVIP has determined that the top guide is susceptible to Irradiation Assisted Stress Corrosion Cracking (IASCC) since the fluence on the top guide exceeds the screening threshold of 5E20 neutrons/square centimeters early in the original 40-year license term. The boiling water reactor (BWR) Top Guide Inspection and Flaw Evaluation Guidelines in Boiling Water Reactor Vessel and Internals Project (BWRVIP) report BWRVIP-26-A were developed considering that fluence exceeds the IASCC threshold. GGNS follows the guidelines in BWRVIP-26-A. Both the MPM and GEH models indicate that fluence on the top guide exceed the IASCC threshold. Therefore, the MPM and GEH results for GGNS are sufficiently conservative for determining the fluence used for screening purposes in the reactor vessel internals program.

b. Background

In the brief analytic uncertainty analysis that follows, the sum of peak reactor pressure vessel (RPV) fluence values from two different fluence methods is discussed. These fluence values are the pre-EPU fluence to 22.99 effective full power years (EFPY) and the EPU fluence for the remaining 31.01 EFPY of a total 54 EFPY (60 years at 90% capacity factor) as follows.

MPM pre-EPU (22.99 EFPY) RPV peak fluence = 1.32E18 n/cm^2

GEH post-EPU (31.01 EFPY) RPV peak fluence = 3.12E18 n/cm^2

Analytic Uncertainty Analysis

In accordance with RG 1.190, both the fluence method used to calculate EPU fluence and that used to calculate EPU fluence are supported by respective analytic uncertainty analyses. The peak reported uncertainty value associated with the MPM pre-EPU fluence calculations is that of the RPV fluence uncertainty: 16%. The maximum uncertainty value associated with GEH EPU fluence calculations is also that of the RPV fluence uncertainty: 19%. Both uncertainty values are reported as relative (percentage) values, just as the uncertainty limit is reported in Regulatory Guide (RG) 1.190. Absolute uncertainty may be determined by combining an absolute calculated fluence value with its associated relative uncertainty, though RG 1.190 does not specify absolute uncertainty limits. The term "relative uncertainty" is used herein to distinguish the value from an absolute uncertainty value.

One approach for combination of uncertainties associated with individual fluence values from the two fluence methods would be a sum of normally distributed correlated random variables. This is the same approach used for determining uncertainty when measured values are used to calculate a function, that is,

$$\sigma_q^2 = \left(\frac{\partial q}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial q}{\partial y}\right)^2 \sigma_y^2 + 2\left(\frac{\partial q}{\partial x}\right)\left(\frac{\partial q}{\partial y}\right)\sigma_{xy},$$

[Ref. 1]

where σ_{xy} is the covariance of variables x and y.

However, for the combination of fluence results, we must apply an error propagation technique to calculated values and not measured values. Unlike measured values which may be derived from a number of measurements and which have unique associated expected values, the expected value of a constant is equal to the constant itself. Therefore the covariance for the variables goes to zero, and the errors in x and y may be treated as independent, and the usual root-sum-of-squares formula for error propagation may be applied, i.e.,

$$\sigma_q^2 = \left(\frac{\partial q}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial q}{\partial y}\right)^2 \sigma_y^2.$$

The maximum total uncertainty is determined by this method as follows.

Given:
MPM pre-EPU (22.99 EFPY) RPV peak fluence w/ 16% uncertainty = 1.32E18 +/- 2.11E17 n/cm^2
GEH post-EPU (31.01 EFPY) RPV peak fluence w/ 19% uncertainty = 3.12E18 +/- 5.92E17 n/cm^2
Combined total fluence = 4.44E18 n/cm^2

Relative uncertainty of combined total fluence is calculated by using the root-sum-of-squares to determine absolute uncertainty, then dividing this by the absolute total fluence value, such as,

$$\frac{\sqrt{(2.11E17)^2+(5.92E17)^2}}{4.44E18} = 14.17\%.$$

The independence of all uncertainty terms between the two fluence methods cannot be proven, as it is known that both fluence analyses involve a 2D flux synthesis / discrete ordinates transport calculation using the same DORT code. Given that independence of all uncertainty terms cannot be proven, a conservative approach may be adopted and the provisional error propagation technique for uncertainties in sums of direct measurements may be used, where in adding quantities, the uncertainties in those quantities are simply added as well [1].

The maximum total uncertainty is determined by this method as follows:

$$\frac{2.11E17 + 5.92E17}{4.44E18} = 18.11\%.$$

As the staff indicated in the issue discussion of the RAI, the uncertainty associated with adding the two fluence values together should be higher than either as a stand-alone calculation. However, since the total combined fluence is also higher than the fluence from the individual stand-alone calculations, the relative uncertainty for the total of fluence is between the relative uncertainty of the individual calculations.

Given that the relative uncertainty values for the pre-EPU and EPU RPV fluence methods are the highest reported uncertainty value associated each method, the preceding example demonstrates that the RG 1.190 relative uncertainty limit of 20% is not exceeded by a combination of pre-EPU and EPU fluence results. Even assuming the GEH uncertainty value of 19% for all combined total fluence values, the RG 1.190 relative uncertainty limit is not exceeded.

References:

1. Taylor, John R. An Introduction to Error Analysis, the Study of Uncertainties in Physical Measurements. Sausalito, CA: University Science Books, 1997, pp. 50,221.

RAI 4.7.3-1a

Background. In its response to RAI 4.7.3-1 dated July 25, 2012, as revised by letter dated September 4, 2012, the applicant provided a list of components included in the TLAA of reactor vessel internals fluence effects. The applicant also provided the 40- and 60-year fluence values for these components and indicated that these values were calculated in accordance with General Electric Licensing Topical Report NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," Revision 2 (GE methodology).

Issue.

a. Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) Section 4.7.3.1.2 states that, for TLAA dispositions pursuant to 10 CFR 54.21(c)(1)(ii), the applicant shall provide a sufficient description of the re-analysis and document the results to show that it is satisfactory for the 60-year period. In RAI 4.7.3-1, the staff requested the applicant to identify the applicable design requirements to show that the results of the re-analysis meet those requirements. The applicant did not provide this information; therefore, the applicant has not shown that the results of the re-analysis are satisfactory for 60 years.

b. SRP-LR Section 4.7.3.1.2 also states that the applicable analysis technique can be the one that is in effect in the plant's current licensing basis (CLB) at the time when the license renewal application is filed. The response to RAI 4.7.3-1 states that the GE methodology was used to calculate the 60-year fluence values for the reactor vessel internals. This 60-year period includes pre-EPU fluence and post-EPU fluence through the period of extended operation. The response to RAI 4.2.1-2 states that the GE methodology was incorporated into the CLB during EPU license amendment approval, which occurred after the license renewal application was filed. As such, it is not clear whether use of the GE methodology for calculating pre-EPU fluence values was consistent with the CLB.

c. The response to RAI 4.7.3-1 states that non-core support structure components (i.e., jet pump beam bolt and shroud head studs) were evaluated in the TLAA of reactor vessel internals fluence effects; however, the updated final safety analysis report (UFSAR) supplement summary description of the TLAA in LRA Section A.2.5.3 does not address the analysis of these components.

Request

a. Provide the applicable design requirements for each component included in the TLAA (e.g., fluence values, bolting preload values). Show that these requirements are met as a result of the re-analysis of fluence effects through the proposed 60 years of plant operation.

b. Justify use of the GE methodology for calculating fluence for the reactor vessel internals pre-EPU.

4.7.3-1a RESPONSE:

a. For austenitic stainless steel components subjected to a neutron fluence level less than 1E21 nvt (E_f > 1 MeV), the base material requires no special consideration above meeting basic American Society of Mechanical Engineers (ASME) Code requirements. For austenitic stainless steel components with less than 5E20 nvt (E_f > 1 MeV), the weld material requires no special consideration, where nvt is fluence, E_f is Energy, and MeV is Million electron volts.

At fluence greater than 1E21 nvt (E_f > 1 MeV) for 304 or 304L base material and 5E20 nvt (E_f > 1 MeV) for 308 or 308L welds, that portion of the component and weld exposed to this greater fluence shall meet the following criteria in addition to ASME Code requirements.

a. For normal and upset conditions:

$$\frac{P_m+Q_m}{E} \le \frac{1}{3}\varepsilon_{unif} \quad \text{and} \quad \frac{P_m+P_b+Q}{E} \le 0.45\ \varepsilon_{unif}$$

b. For emergency conditions:

$$\frac{P_m+Q_m}{E} \le \frac{1}{2}\varepsilon_{unif} \quad \text{and} \quad \frac{P_m+P_b+Q}{E} \le 0.67\ \varepsilon_{unif}$$

c. For faulted conditions:

$$\frac{P_m+Q_m}{E} \le \frac{2}{3}\varepsilon_{unif} \quad \text{and} \quad \frac{P_m+P_b+Q}{E} \le 0.90\ \varepsilon_{unif}$$

Where ε_{unif} is the uniform elongation which has a value of 0.5 percent for these conditions at 550°F and has a value of 0.6 percent for these conditions at 75°F. Between 75°F and 550°F, the value of ε_{unif} may be linearly interpolated between 0.5 and 0.6 percent.

308 or 308L welds subjected to fluences greater than 5E20 nvt (E_f > 1 MeV), shall either be limited to a maximum stress of 5,000 psi resulting from any operating and accident load or be full penetration welds with a minimum quality of 0.9 as determined by Table NG-3352-1 of ASME Code.

In addition to meeting ASME Code threaded structural fastener requirements, the bolted joint shall be designed to provide for relaxation (thermal and irradiation) using the maximum expected fluence values.

The allowable strains and required bolt clamping loads for RPV internals due to fluence effects at 60 years of plant operation are listed in the following table. The calculated strains at 60 years of plant operation are less than the allowable strains. The calculated bolt clamping loads are more than the required bolt clamping loads. Hence all the design specification requirements are met for RPV internal components at 60 years of plant operation.

Key Results Affected By Fluence

[refer to Enclosure 1 Table 3.5-1 "Key Results Affected By Fluence"]

Core Support Structure (CSS) components were designed in accordance with ASME B&PV Code, Section III, Division 1 as stated in General Electric (GE) design specification 22A4052, Core Support Structure. Non-CSS components are 'non-ASME code' components. However, ASME Code criteria were used for non-CSS components as guidelines wherever applicable, consistent with applicable design basis.

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

Irradiation does not affect stress criteria, but affects strain, material ductility and relaxation. Hence, the non-CSS components of jet pump beam bolts and shroud head studs were also evaluated for performance and functional requirements, and were found acceptable. The other non-CSS components are either away from the active fuel in the reactor core such that they are not impacted by irradiation, or changes in strain, ductility and relaxation do not impact their structural adequacy and functionality.

b. Design basis pre-EPU fluence values were used in total end-of-life fluence calculations for the reactor vessel internals where available. In cases where design basis pre-EPU fluence values were unavailable, pre-EPU flux/fluence values were derived from the GEH-calculated EPU flux distribution.

GEH-derived pre-EPU flux values were derived from EPU values calculated per the NRC-approved GEH fluence methodology. The flux synthesis method used to generate the EPU values which serve as the bases for these calculations in all cases assumed the peak azimuthal flux from the (r,θ) flux distribution to bound the flux at the locations of interest.

For certain vessel internals components with radial spans within the shroud inner radius (IR), synthesized flux was calculated using the maximum (r,θ) flux distribution value. For these components, pre-EPU flux values were derived by multiplying the ratio of original licensed thermal power (OLTP) flux to EPU flux for the shroud H4 weld with calculated EPU flux values. The ratio is taken between EPU and OLTP to maintain consistency with available design basis pre-EPU calculated fluences, which assumed that the pre-EPU cycles accumulated 22.99 EFPY normalized to the original power rating of 3833 megawatt thermal (MWth). For components whose assumed radial locations are within the shroud IR, the H4 EPU/OLTP flux ratio is considered the best reasonable scalar to estimate pre-EPU flux values.

For all other evaluated vessel internals components, pre-EPU flux was derived by applying the OLTP/EPU power adjustment constant of (3833/4408) to calculated EPU flux values. Again, the OLTP/EPU ratio is used to maintain consistency with the assumption that the pre-EPU cycles have cumulative life of 22.99 EFPY normalized to the original power rating of 3833 MWth. Use of a power adjustment factor in this case is considered the best reasonable scalar to estimate pre-EPU flux values.

Enclosure 2 to

GNRO-2013/00003

1-2OR45T-13 GEH Responses to Grand Gulf License Renewal RAIs
GEH Non-proprietary Information - Class I (Public)

ENCLOSURE 2

1-2OR45T-13

GEH Responses to Grand Gulf License Renewal RAIs

GEH Non-proprietary - Class I (Public)

NON-PROPRIETARY NOTICE

This is a non-proprietary version of the Enclosure 1 of 1-2OR45T-13 which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

RAI 4.2.1-2b:

Background.

In its License Renewal Application, the applicant used a reactor vessel fluence estimate that was obtained by combining a pre-extended power uprate (EPU) fluence value obtained from MPM Technologies calculations to a post-EPU fluence value obtained from General Electric Hitachi (GEH).

By letter dated October 15, 2012, the applicant responded to RAI 4.2.1-2a that, in part, addressed the discrepancies between the pre-EPU and post-EPU flux values for the core shroud welds (H1, V1, V2, V3, and V4 welds in the top portion of the core shroud). In the applicants analysis for these weld locations, the pre-EPU peak flux values are greater than the post-EPU peak flux values approximately by three orders of magnitude (i.e., approximately 1.0E10 n/cm2-s versus 1.0E07 n/cm2-s). The core shroud neutron flux values are described in the applicant's response, dated July 25, 2012, to RAI 4.2.1-2.

In addition, RAI4.2.1-2a addressed the issue regarding the combined analytic uncertainty associated with the two different methods for the pre-EPU and post-EPU fluence calculations (i.e., MPM method and GEH method, respectively).

Issue

The applicant's response to RAI 4.3.1-2a states that the MPM approach used a bounding-case flux at the core shroud weld locations in the top portion of the core shroud shell. However, the applicant's response does not provide specific information to justify why the flux values obtained from the MPM method are bounding-case flux values for the core shroud weld locations.

In addition, the applicant's response indicates the following with respect to the combined analytic uncertainty:

- *The applicant acknowledged that proving the independence of all uncertainty terms in both fluence calculations is not possible.*
- *The applicant stated that because the independence of the uncertainty values cannot be proven, relative uncertainties are considered instead.*
- *The applicant expressed total uncertainty as a "relative uncertainty' combination, which is, in actuality, a weighted average of the uncertainties associated with both methods.*

The staff noted the following issues with the applicant's response:

- *The applicants approach is not a valid or acceptable way to perform an analytic uncertainty analysis because it is not consistent with Regulatory Guide (RG) 1.190.*
- *The errors associated with both fluence calculation methods will propagate when the two fluence calculational methods are combined.*
- *The uncertainty associated with adding the two fluence values together should be higher than either as a stand-alone calculation.*

<u>Request</u>

a. *Provide additional information to clarify why the flux values obtained from the MPM method are bounding-case flux values for the core shroud weld locations (i.e., H1, V1, V2, V3, and V4 welds). As part of the response, clarify which locations are actually associated with the bounding-case flux values and why the bounding-case flux values have sufficient conservatisms as bounding-case values.*

   Entergy Response

b. *Provide an analytic uncertainty analysis for the combined fluence values, consistent with RG 1.190 that states that each of the fluence methods be supported by an analytic uncertainty analysis. Alternatively, provide reactor vessel neutron fluence values that were calculated using a single, NRC-approved method that has a valid and accepted analytic uncertainty analysis, and reconfirm the validity of the related neutron embrittlement time-limited aging analysis (TLAAs) in light of the updated fluence values.*

<u>Background</u>

In the brief analytic uncertainty analysis that follows, the sum of peak RPV fluence values from two different fluence methods is discussed. These fluence values are the pre-EPU fluence to 22.99 EFPY and the EPU fluence for the remaining 31.01 EFPY of a total 54 EFPY (60 years at 90% capacity factor) as follows.

MPM pre-EPU (22.99 EFPY) RPV peak fluence = 1.32E18 n/cm^2

GEH post-EPU (31.01 EFPY) RPV peak fluence = 3.12E18 n/cm^2

<u>Analytic Uncertainty Analysis</u>

In accordance with RG 1.190, both the fluence method used to calculate pre-EPU fluence and that used to calculate EPU fluence are supported by respective analytic uncertainty analyses. The peak reported uncertainty value associated with the MPM pre-EPU fluence calculations is that of the RPV fluence uncertainty: 16%. The maximum uncertainty value associated with GEH EPU fluence calculations is also that of the RPV fluence uncertainty: 19%. Both uncertainty values are reported as relative (percentage) values, just as the uncertainty limit is reported in RG 1.190. Absolute uncertainty may be determined by combining an absolute calculated fluence value with its associated relative uncertainty, though RG 1.190 does not specify absolute uncertainty limits. The term "relative uncertainty" is used herein to distinguish the value from an absolute uncertainty value.

One approach for combination of uncertainties associated with individual fluence values from the two fluence methods would be a sum of normally distributed correlated random variables. This is the same approach used for determining uncertainty when measured values are used to calculate a function, that is,

$$\sigma_q^2 = \left(\frac{\partial q}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial q}{\partial y}\right)^2 \sigma_y^2 + 2\left(\frac{\partial q}{\partial x}\right)\left(\frac{\partial q}{\partial y}\right)\sigma_{xy}$$

(Reference 1)

where σ_{xy} is the covariance of variables x and y.

However, for the combination of fluence results, we must apply an error propagation technique to calculated values and not measured values. Unlike measured values which may be derived from a number of measurements and which have unique associated expected values, the expected value of a constant is equal to the constant itself. Therefore the covariance for the variables goes to zero, and the errors in x and y may be treated as independent, and the usual root-sum-of-squares formula for error propagation may be applied, i.e.,

$$\sigma_q^2 = \left(\frac{\partial q}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial q}{\partial y}\right)^2 \sigma_y^2$$

The maximum total uncertainty is determined by this method as follows.

Given: MPM pre-EPU (22.99 EFPY) RPV peak fluence w/ 16% uncertainty = 1.32E18 +/- 2.11E17 n/cm^2

GEH post-EPU (31.01 EFPY) RPV peak fluence w/ 19% uncertainty = 3.12E18 +/- 5.92E17 n/cm^2

Combined total fluence = 4.44E18 n/cm^2

Relative uncertainty of combined total fluence is calculated by using the root-sum-of-squares to determine absolute uncertainty, then dividing this by the absolute total fluence value, such as,

$$\frac{\sqrt{(2.11\text{E}17)^2 + (5.92\text{E}17)^2}}{4.44\text{E}18} = 14.2\%$$

The independence of all uncertainty terms between the two fluence methods cannot be proven, as it is known that both fluence analyses involve a 2D flux synthesis / discrete ordinates transport calculation using the same DORT code. Given that independence of all uncertainty terms cannot be proven, a conservative approach may be adopted and the provisional error propagation technique for uncertainties in sums of direct measurements may be used, where in adding quantities, the uncertainties in those quantities are simply added as well (Reference 1).

The maximum total uncertainty is determined by this method as follows:

$$\frac{2.11\text{E}17 + 5.92\text{E}17}{4.44\text{E}18} = 18.1\%$$

Given that the relative uncertainty values for the pre-EPU and EPU RPV fluence methods are the highest reported uncertainty values associated with each method, the preceding example demonstrates that the RG 1.190 relative uncertainty limit of 20% is not exceeded by a combination of pre-EPU and EPU fluence results. Even assuming the GEH uncertainty value of 19% for all combined total fluence values, the RG 1.190 relative uncertainty limit is not exceeded.

References:

1. Taylor, John R. An Introduction to Error Analysis, the Study of Uncertainties in Physical Measurements. Sausalito, CA: University Science Books, 1997, pp. 50,221.

Non-proprietary Information

RAI 4.7.3-1a:

Background.

In its response to RAI4.7.3-1 dated July 25, 2012, as revised by letter dated September 4, 2012, the applicant provided a list of components included in the TLAA of reactor vessel internals fluence effects. The applicant also provided the 40-and 60-year fluence values for these components and indicated that these values were calculated in accordance with General Electric Licensing Topical Report NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations;' Revision 2 (GE methodology).

Issue.

a. *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) Section 4.7.3.1.2 states that, for TLAA dispositions pursuant to 10 CFR 54.21 (c)(1)(ii), the applicant shall provide a sufficient description of the re-analysis and document the results to show that it is satisfactory for the 60-year period. In RAI4.7.3-1, the staff requested the applicant to identify the applicable design requirements to show that the results of the re-analysis meet those requirements. The applicant did not provide this information; therefore, the applicant has not shown that the results of the re-analysis are satisfactory for 60 years.*

b. *SRP-LR Section 4.7.3.1.2 also states that the applicable analysis technique can be the one that is in effect in the plants current licensing basis (CLB) at the time when the license renewal application is filed. The response to RAI4.7.3-1 states that the GE methodology was used to calculate the 60-year fluence values for the reactor vessel internals. This 60-year period includes pre-EPU fluence and post-EPU fluence through the period of extended operation. The response to RAI 4.2.1-2 states that the GE methodology was incorporated into the CLB during EPU license amendment approval, which occurred after the license renewal application was filed. As such, it is not clear whether use of the GE methodology for calculating pre-EPU fluence values was consistent with the CLB.*

c. *The response to RAI4.7.3-1 states that non-core support structure components (Le., jet pump beam bolt and shroud head studs) were evaluated in the TLAA of reactor vessel internals fluence effects; however, the updated final safety analysis report (UFSAR) supplement summary description of the TLAA in LRA Section A.2.5.3 does not address the analysis of these components.*

Request

a. *Provide the applicable design requirements for each component included in the TLAA (e.g., fluence values, bolting preload values). Show that these requirements are met as a result of the re-analysis of fluence effects through the proposed 60 years of plant operation.*

For austenitic stainless steel components subjected to a neutron fluence level less than 1E21 nvt ($E_f > 1$ MeV), the base material requires no special consideration above meeting basic ASME Code requirements. For austenitic stainless steel components with less than 5E20 nvt ($E_f > 1$ MeV), the weld material requires no special consideration, where nvt is fluence, E_f is Energy, and MeV is Million electron volts.

At fluence greater than 1E21 nvt ($E_f > 1$ MeV) for 304 or 304L base material and 5E20 nvt ($E_f > 1$ MeV) for 308 or 308L welds, that portion of the component and weld exposed to this greater fluence shall meet the following criteria in addition to ASME Code requirements.

a. For normal and upset conditions:

$$\frac{P_m + Q_m}{E} \le \frac{1}{3}\varepsilon_{unif} \quad \text{and} \quad \frac{P_m + P_b + Q}{E} \le 0.45\varepsilon_{unif}$$

b. For emergency conditions:

$$\frac{P_m + Q_m}{E} \le \frac{1}{2}\varepsilon_{unif} \quad \text{and} \quad \frac{P_m + P_b + Q}{E} \le 0.67\varepsilon_{unif}$$

c. For faulted conditions:

$$\frac{P_m + Q_m}{E} \le \frac{2}{3}\varepsilon_{unif} \quad \text{and} \quad \frac{P_m + P_b + Q}{E} \le 0.90\varepsilon_{unif}$$

Where ε_{unif} is the uniform elongation which has a value of 0.5 percent for these conditions at 550°F and has a value of 0.6 percent for these conditions at 75°F. Between 75°F and 550°F, the value of ε_{unif} may be linearly interpolated between 0.5 and 0.6 percent.

308 or 308L welds subjected to fluences greater than 5E20 nvt ($E_f > 1$ MeV), shall either be limited to a maximum stress of 5,000 psi resulting from any operating and accident load or be full penetration welds with a minimum quality of 0.9 as determined by Table NG-3352-1 of ASME Code.

In addition to meeting ASME Code threaded structural fastener requirements, the bolted joint shall be designed to provide for relaxation (thermal and irradiation) using the maximum expected fluence values.

The allowable strains and required bolt clamping loads for RPV internals due to fluence effects at 60 years of plant operation are listed in Table 3.5-1. The calculated strains at 60 years of plant operation are less than the allowable strains. The calculated bolt clamping loads are more than the required bolt clamping loads. Hence all the design specification requirements are met for RPV internal components at 60 years of plant operation.

Table 3.5-1 Key Results Affected By Fluence

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Core Support Structure (CSS) components were designed in accordance with ASME B&PV Code, Section III, Division 1 as stated in GE design specification 22A4052, Core Support Structure. Non-CSS components are 'non-ASME code' components. However, ASME Code criteria were used for non-CSS components as guidelines wherever applicable, consistent with applicable design basis.

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

Irradiation does not affect stress criteria, but affects strain, material ductility and relaxation. Hence, the non-CSS components of jet pump beam bolts and shroud head studs were also evaluated for performance and functional requirements, and were found acceptable. The other non-CSS components are either away from the active fuel in the reactor core such that they are not impacted by irradiation, or changes in strain, ductility and relaxation do not impact their structural adequacy and functionality.

b. Justify use of the GE methodology for calculating fluence for the reactor vessel internals pre-EPU.

Design basis pre-EPU fluence values were used in total end-of-life fluence calculations for the reactor vessel internals where available. In cases where design basis pre-EPU fluence values were unavailable, pre-EPU flux/fluence values were derived from the GEH-calculated EPU flux distribution.

GEH-derived pre-EPU flux values were derived from EPU values calculated per the NRC-approved GEH fluence methodology. The flux synthesis method used to generate the EPU values which serve as the bases for these calculations in all cases assumed the peak azimuthal flux from the (r,θ) flux distribution to bound the flux at the locations of interest.

For certain vessel internals components with radial spans within the shroud inner radius, synthesized flux was calculated using the maximum (r,θ) flux distribution value. For these components, pre-EPU flux values were derived by multiplying the ratio of original licensed thermal power (OLTP) flux to EPU flux for the shroud H4 weld with calculated EPU flux values. The ratio is taken between EPU and OLTP to maintain consistency with available design basis pre-EPU calculated fluences, which assumed that the pre-EPU cycles accumulated 22.99 EFPY normalized to the original power rating of 3833 MWth. For components whose assumed radial locations are within the shroud IR, the H4 EPU/OLTP flux ratio is considered the best reasonable scalar to estimate pre-EPU flux values.

For all other evaluated vessel internals components, pre-EPU flux was derived by applying the OLTP/EPU power adjustment constant of (3833/4408) to calculated EPU flux values. Again, the OLTP/EPU ratio is used to maintain consistency with the assumption that the pre-EPU cycles have cumulative life of 22.99 EFPY normalized to the original power rating of 3833 MWth. Use of a power adjustment factor in this case is considered the best reasonable scalar to estimate pre-EPU flux values.

Enclosure 3 to

GNRO-2013/00003

1-2OR45T-13 Affidavit

ENCLOSURE 3

1-2OR45T-13

Affidavit

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Linda C. Dolan, state as follows:

(1) I am the Manager, Regulatory Compliance of GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) that is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter, 1-2OR45T-13, "NRC Grand Gulf Nuclear Station License Renewal RAIs," dated January 17, 2013. GEH proprietary information in Enclosure 1 which is entitled "GEH Responses to Grand Gulf License Renewal RAIs – GEH Proprietary Information - Class III (Confidential)" is identified by a dark red dotted underline inside double square brackets, [[This sentence is an example.{3}]]. Tables containing GEH proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act (FOIA), 5 U.S.C. Sec. 552(b)(4), and the Trade Secrets Act, 18 U.S.C. Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 as decided in Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary or confidentiality agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains criteria underlying the analysis performed by GEH to support the Grand Gulf License Renewal submittal. Development of the methodology and supporting analysis, techniques, and information and their application for the design, modification, and processes were achieved at a significant cost GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 17th day of January 2013.

Linda C. Dolan

Linda C. Dolan
Manager, Regulatory Compliance
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd
Wilmington, NC 28401
Linda.dolan@ge.com