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U.S. Nuclear Regulatory Commission
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Subject: **Response to Portions of NRC Requests for Additional Information Letter 67 Related To ESBWR Design Certification Application -- DCD Chapter 3 – Design of Structures, Components, Systems, and Equipment - RAI 3.9-148 and RAI 3.9-149 S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to portions of the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC Letter dated October 10, 2006, Reference 1 (RAI 3.9-148), and the response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC e-mail dated May 24, 2007, Reference 2 (RAI 3.9-149 S01). The previous response (RAI 3.9-149) was submitted via Reference 3 in response to Reference 1.

Please note that RAI 3.9-149 S01 was received via a separate e-mail from the NRC (Chandu Patel).

If you have any questions or require additional information, please contact me.

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

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References:

1. MFN 06-378, Letter from the U.S. Nuclear Regulatory Commission to David H. Hinds, *Request for Additional Information Letter No. 67 Related To ESBWR Design Certification Application*, dated October 10, 2006.
2. E-mail from Chandu Patel (NRC), Supplement 1 to RAI 3.9-149, dated May 24, 2007.
3. MFN 07-238, Letter from Jim Kinsey to the U.S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application - Reactor Internal Structures - RAI Number 3.9-149*, dated April 30, 2007.

Enclosures:

1. MFN 07-652, Response to Portion of NRC Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application Mechanical Systems and Components - RAI 3.9-148, and response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC e-mail – RAI 3.9-149 S01

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Enclosure 1

MFN 07-652

Response to Portion of NRC Request for

Additional Information Letter No. 67

Related to ESBWR Design Certification Application

Mechanical Systems and Components

RAI Numbers 3.9-148 and 3.9-149 S01

NRC RAI 3.9-148

As indicated in DCD Tier 2, Section 3.9.5.4, GE stated that the design and construction of the core support structures are in accordance with the ASME Code, Subsection NG. GE is requested to identify the specific paragraphs of Subsection NG that are followed for the design and construction of the core support structures. In addition, in Tables 3.9-4 through 3.9-7 of DCD Tier 2, GE provides the stress, deformation, and fatigue criteria for safety-related reactor internals (except core support structures), which are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field experience and testing. GE is requested to: (1) identify which specific paragraphs of Subsection NG from which these criteria are derived, or (2) if other than the ASME Code is used, identify and justify the other criteria (based on manufacturers' standards or empirical methods) that are used as the basis to develop the stress, deformation, and fatigue criteria for safety-related reactor internals.

GEH Response

Core support structures are designed and built to ASME, Section III, Subsection NG. The stress analysis being used is an elastic analysis method that is most commonly performed on the reactor core support structures in accordance with ASME Section III, Subsection NG, Sub-article NG-3200 for Service Conditions A&B, C, and D; and Section III, Appendix F as applicable for Service Level D condition. An inelastic analysis method is also used for a postulated blowout of a CRD Housing caused by a weld failure as discussed in Section 3.9.1.4.

The ASME Code, Section III does not set specific stress limits for Reactor Internal Structures. Per NG-1122*(c)* "The Certificate Holder shall certify that the construction of all internal structures is such as not to affect adversely the integrity of the core support structure." To ensure that the internal structures meet this requirement, the Safety Factors for Level A Level B, Level C and Level D as shown in DCD 3.9.5.are selected so that the calculated stress levels will meet the stress limits for Core Support Structures given in Article NG-3200. For example, in Table 3.9-5 of the DCD, Tier 2, the ratios between the elastic evaluated primary stresses, PE, and the permissible primary stresses, PN shall be :

$$PE/PN \leq 2.25/SF_{min}$$

Applying a Safety Factor of 2.25 to Levels A and B makes $PE/PN \leq 1$ or $PE \leq 1$ PN. Where PE is the elastic primary stress and PN equals the Code limit of 1 S_m for Level A and B. The limit given in Fig. NG-3221-1 is met.

Similarly for Level C, $SF_{min} = 1.5$. Hence, $PE/PN \leq 2.25/1.5$ or $PE \leq 1.5$ PN, which meets the requirements of Fig. NG-3224-1

For Level D, $SF_{min} = 1.125$. Hence, $PE/PN \leq 2.25/1.125$ or $PE \leq 2.0$ PN which is more conservative than the 2.4 S_m limit set by Appendix F of the ASME Code, Section III.

Similarly, when using the largest lower bound limit load, CL, the Permissible load LP :

$LP/CL \leq 1.5/SF_{min}$. When $SF_{min} = 2.25$, $LP = 0.667\ CL$, which is consistent with NG-3228.2 and Fig. NG-3221-1.

When using the conventional ultimate strength at temperature, US, as a limit, the elastic evaluated primary stress, PE : $PE/US \leq 0.75/SF_{min}$. When $SF_{min} = 2.25$,

$PE \leq 0.33\ US$, that meets the stress intensity criterion of ASME Section II, Part D, Appendix 2 110*(b)*.

Moreover, the criterion shown in Tables 3.9-4 through 3.9-7 is developed from Subsection NG of the ASME Code. Per NG-3224.6 the deformation limit can be derived from the ultimate load determined by testing. The elastic limit therefore can be determined as a specified fraction of this load. Per NG-3228.4, NG-3224.1 (e), and NG-3225, this fraction is .44, .6 and .88 for service levels A or B, C and D respectively.

Note: In Table 3.9-4 entitled, Deformation Limit for Safety Class Reactor Internal Structures Only, the footnote (**) to equation b of Table 3.9-4 shall be changed to read: "Equation b will not be used unless supporting data are provided to the NRC." DCD impact has been addressed in GEH's Response to RAI 3.9-149 S01.

DCD Impact

No DCD changes will be made in response to this RAI.

For historical purposes, the original text of RAI 3.9-149 and GEH response is included preceding the supplemental response. To prevent confusion, any original attachments or DCD mark-ups are not included.

NRC RAI 3.9-149

Table 3.9-4 of the DCD, Tier 2, provides deformation limits for safety class reactor internal structures. GE is requested to provide the technical basis for the General Limits listed in the table.

GE Response

The deformation limit of $0.9/SF_{min}$ given in Table 3.9-4 a. of DCD, Tier 2 is determined as follows:
Per the ASME Code, Section II, Part D, Appendix I, the allowable stress intensity value, S_m, for austenitic stainless steel is 90% of minimum yield strength at temperature. Based on experimental data from the industry, the minimum strain, ε, before yield of irradiated stainless steel is selected. Using these values and considering the minimum safety factors, SF_{min}, from Chapter 3.9.5.4 of DCD, Tier 2, the maximum permissible deformation can generally be specified as:

$$(P + Q)/E \leq 0.9/SF_{min} \cdot \varepsilon$$

Where:

- P = Primary stress
- Q = Secondary stress
- E = Young's modulus
- ε = minimum strain before yield

For ASME III, Service Level A and B loads, the maximum permissible deformation with $SF_{min} = 2.25$ would be:

$$(P + Q)/E \leq (0.9/2.25) \cdot \varepsilon = 0.4 \cdot \varepsilon,$$

likewise:

$$(P + Q)/E \leq (0.9/1.5) \cdot \varepsilon = 0.6 \cdot \varepsilon \text{ for Service Level C loads}$$

and

$$(P + Q)/E \leq (0.9/1.125) \cdot \varepsilon = 0.8 \cdot \varepsilon \text{ for Service Level D loads}$$

These maximum permissible deformation limits and the minimum strain value, ε, are specified in the reactor internals design specification.

When experimental data from the actual material are used, the general deformation limit $1.00/SF_{min}$ as shown in Table 3.9-4 b. may be used.

DCD Impact

No DCD changes will be made in response to this RAI.

NRC RAI 3.9-149 S01

RAI 3.9-149 S01: Comment on response to RAI 3.9-149 (MFN 07-238):

The its response to RAI 3.9-149 in a letter dated April 30, 2007, the applicant states that, according to ASME Code, Section II, Part D, Appendix I, the allowable stress intensity value, Sm, for austenitic stainless steel is 90% of the minimum yield strength at temperature. The applicant has selected the minimum strain, e, just before yielding of irradiated stainless steel to represent the strain corresponding minimum yield strength at temperature. The applicant states that the magnitude of the minimum strain, e, is based on experimental data from industry.

The applicant defines the deformation limits in terms of minimum strain, e, and the safety factors, SFmin, defined in Section 3.9.5.4 of ESBWR DCD, Tier 2, Rev. 1, January 2006. The deformation limits can be expressed as,

$(P + Q)/E \leq (0.9/SFmin) . e$

For Service Levels A to D, according to Section 3.9.5.4, safety factors, SFmin, vary from 2.25 to 1.125.

In response to RAI 3.9-149, the applicant further states that when experimental data from the actual material are used, the general deformation limit 1.00/SFmin instead of 0.9/SFmin, as shown in Table 3.9-4 (b) may be used. The staff needs the following additional information to complete the review:

(a) The applicant is requested to provide a reference for the industry data for irradiated stainless steel as mentioned in its response. In addition, the applicant is requested to provide a summary of this industry data, especially the neutron fluence and irradiation temperature for the irradiated steel considered here. Also, please provide the end-of-the-life neutron fluence for the vessel internals that will be subject to deformation limits.

(b) The applicant is requested to provide technical basis for the safety factors defined in Section 3.9.5.4 of ESBWR DCD, Tier 2, Rev. 1, January 2006.

(c) The applicant is requested to explain the increase in the general deformation limit from 0.9/SFmin to 1.0/SFmin when experimental data from the actual material are used. The applicant is also requested to identify any codes or standards that support such increase in the general deformation limit.

GEH Response

(a) Presently, GEH has no plans to perform any of the Reactor Internal Structures qualification by experimental method (per equation b of Table 3.9-4). If equation b (of

Table 3.9-4) is to be used, GEH will provide all the supporting data to NRC for approval prior to its use. GEH would select the experimental data for irradiated stainless steel from either the internal BWR Materials Property Handbook, or other industrial data that may be available at the time.

The end-of-life neutron fluence for vessel internals varies based on the time-integrated flux calculated for the region where the components are located. For ESBWR, neutron fluence for vessel internals are represented by significant internal components such as the shroud, top guide, and core plate; the data and plots can be found in Section 4.0 of the ESBWR Neutron Fluence Evaluation. This document contains GE proprietary data and is available for the NRC to review at the GE Licensing Offices in Washington, D.C, or Wilmington, NC.

(b) See the response to RAI 3.9-148 for the technical basis of the safety factors defined in DCD Tier 2, Subsection 3.9.5.4.

(c) The general deformation limit can only be increased if sufficient experimental data that are approved by the NRC are available. Note (**) for equation b of Table 3.9-4 is being changed to read: "Equation b will not be used unless supporting data are provided to the NRC." ASME B&PV Code, Section III, Subsection NG supports the increase in the deformation limit. The code states that the deformation limits are defined by the design specification.

DCD Impact

DCD Tier 2, Table 3.9-4, Note (**) will be revised as noted in the attached markup.

Table 3.9-4

Deformation Limit for Safety Class Reactor Internal Structures Only

	Either One Of (Not Both)		General Limit
a.	$\frac{\text{Permissible deformation, DP}}{\text{Analyzed deformation causing loss of function, DL}}$	$\leq$	$\frac{0.90}{SF_{min}}$
b.**	$\frac{\text{Permissible deformation, DP}}{\text{Experimental deformation causing loss of function, DE}}$	$\leq$	$\frac{1.00}{SF_{min}}$

where:

DP = Permissible deformation under stated conditions of Service Levels A, B, C or D (normal, upset, emergency or fault).

DL = Analyzed deformation which could cause a system loss of function*.

DE = Experimentally determined deformation which could cause a system loss of function.

SF_{min} = Minimum safety factor (refer to Subsection 3.9.5.4).

Notes:

* "Loss of Function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they may be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are control rod drive alignment and clearances for proper insertion, or excess leakage of any component.

** Equation b will not be used unless supporting data are provided to the NRC.