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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 122 Related to ESBWR Design Certification Application - Pressure and Temperature Limits Report - RAI Number 5.3-7 S02**

Enclosure 1 contains the GE Hitachi Nuclear Energy (GEH) response to the subject NRC RAI originally transmitted via the Reference 1 letter and supplemented by an NRC request for clarification in Reference 2.

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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NRO

References:

1. MFN 06-093, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 14 Related to ESBWR Design Certification Application*, March 22, 2006
2. MFN 07-659, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request for Additional Information Letter No. 122 Related to ESBWR Design Certification Application*, December 6, 2007

Enclosure:

1. MFN 06-128 Supplement 2 - Response to Portion of NRC Request for Additional Information Letter No. 122 Related to ESBWR Design Certification Application - Pressure and Temperature Limits Report - RAI Number 5.3-7 S02

cc:	AE Cubbage	USNRC (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRF	0000-0080-4124

Enclosure 1

MFN 06-128 Supplement 2

Response to Portion of NRC Request for Additional Information Letter No. 122 Related to ESBWR Design Certification Application

Pressure and Temperature Limits Report

RAI Number 5.3-7 S02

NRC RAI 5.3-7 S02:

In Revision 4 of the DCD Tier 2, Section 5.3.1.5, the applicant stated, "An updated Pressure and Temperature Limits Report (PTLR) is developed in accordance with the Technical Specifications." The applicant has not yet submitted the PTLR for NRC review and approval, thus, the statement referring to an update is not appropriate at this time. The applicant needs to clearly identify who (COL Applicant or GEH) will prepare the PTLR and when it will be submitted for NRC review and approval. The staff requests that GEH revise the DCD Sections 5.3.1.5 and 5.3.4 accordingly to reflect this change including adding a COL Information Item if the PTLR will not be submitted by GEH as part of the design certification.

GEH Response:

The COL Holder, in accordance with DCD Tier 2, Chapter 16, Technical Specification 5.6.4, will furnish the plant specific reactor pressure vessel pressure/temperature (P/T) curves in a Pressure and Temperature Limits Report (PTLR) after shipment of the reactor pressure vessel, since the reactor pressure vessel material properties are known at that time.

Since Technical Specification 5.6.4 requires the PTLR be provided to the NRC, no COL action item is necessary.

DCD Impact:

DCD Tier 2 Subsections 5.3.1.5 and 5.3.2.1 will be revised as shown in the attached markup.

DCD Tier 2 Subsection 5.3.4 will not be revised since a COL action item is unnecessary.

Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed under Regulatory Guide 1.71 in Subsection 5.2.3.4.2 of this report.

Regulatory Guide 1.99: Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper shelf energy are made in accordance with the requirements of Regulatory Guide 1.99.

Regulatory Guide 1.37: Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

The cleaning of systems and components on the site during and at the completion of construction is accomplished to written procedures, which ensures both cleanliness and that the components are not exposed to materials or practices which may degrade their performance. For components containing stainless steel, the procedures shall comply with Regulatory Guide 1.37. The procedures prohibit contact with low melting point compounds, substances which are known to cause stress corrosion cracking or which can release, in any manner, substances that can cause such problems. In addition, there are controls placed on the use of grinding wheels and wire brushes, which assures that they cannot introduce degrading materials either through prior usage or through their materials of construction. In this context, degradation includes stress corrosion cracking. Controls also control introduction of unnecessary dirt and require control of dirt producing processes such as welding or grinding including prompt cleaning.

5.3.1.5 Fracture Toughness

Compliance with 10CFR50, Appendix G

Appendix G of 10CFR50 is interpreted for Class 1 primary coolant pressure boundary components of the ESBWR design and complied with as discussed in Methods of Compliance below and Subsection 5.3.2. The specific temperature/pressure limits for the operation of the reactor (Figures 5.3-1 and 5.3-2) are based on 10CFR50 Appendix G, Paragraph IV, A.2. ~~An updated Pressure and Temperature Limits Report (PTLR) is developed in accordance with the Technical Specifications~~ The plant specific pressure/temperature limits will be prepared using actual reactor pressure vessel materials properties that will be submitted in a Pressure and Temperature Limits Report (PTLR) by the COL Holder. The COL Holder, in accordance with the Technical Specification (Chapter 16, Section 5.6.4), will furnish the PTLR for NRC review after shipment of the reactor pressure vessel since the reactor pressure vessel material properties are known at that time.

- Material Test Coupons and Test Specimens (Appendix G III-A)
- 10 CFR Part 50, §50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," as it relates to fracture toughness criteria for PWRs relevant to pressurized thermal shock events is not applicable to the ESBWR; and
- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," as it relates to material testing and fracture toughness.

The specific criteria which meet the relevant requirements are as presented in the following subsections.

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figures 5.3-1 and 5.3-2 are representative for the ESBWR. They are based on the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99.

The vessel flange, RPV head and flange areas, feedwater nozzles, bottom head and the core beltline areas were evaluated using the material initial RT_{NDT} data from the RPV specification. The operating limit curves are based on the most limiting locations. The pressure/temperature limits are based on flaw sizes specified in Paragraph G-2120 of ASME Section III, Appendix G. The maximum through wall temperature gradient from continuous heating or cooling at 55.6°C (100°F) per hour was considered. The safety factors applied were as specified in ASME Section III, Appendix G.

To calculate the adjusted reference temperature (accounting for the effects of irradiation in the vessel beltline region), the copper and nickel specification limits were used in combination with the peak fluence values and the methodology of RG 1.99, Revision 2. This is considered conservative since the actual RT_{NDT} values and chemical composition are normally much lower than the ones specified. Margins for the adjusted reference temperature calculation are as defined in RG 1.99 Revision 2.

For each individual component (e.g., main steam nozzle), a finite element model is used to determine the stresses (pressure and thermal) for the transient events for normal and upset conditions. These stresses are then used to determine the applied K_I for each transient. The most limiting transient K_I for a given pressure and temperature is then compared to the minimum required K_I (note that the minimum temperature limits of 10 CFR 50 Appendix G also apply). The minimum required K_I is based upon the limiting RT_{NDT} of the materials for the component (determined per above), and calculated using the methodology of ASME Section III, Appendix G. For the pressure test condition, a safety factor of 1.5 is applied to K_{Ip} (K_I from primary membrane and bending stresses). For the core not critical and core critical conditions, a factor of 2.0 is applied to K_{IP}). The RT_{NDT} of the vessel materials are determined in accordance with the ASME Section III, Subsection NB-2320, and the requirements are listed in Table 5.3-1.

Temperature Limits for Boltup

Minimum flange and fastener temperatures of RT_{NDT} plus 33°C (60°F) are required for tensioning at preload condition and during detensioning. As shown in Table 5.3-1, this is higher than that calculated in accordance with the methods described in ASME Section III, Appendix G.

Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Minimum flange and fastener temperatures of RT_{NDT} plus 33°C (60°F) are required for tensioning at preload condition and during detensioning. As shown in Table 5.3-1, this is higher than that calculated in accordance with the methods described in ASME Section III, Appendix G.

Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Representative pressure versus temperature limits for preservice and inservice tests when the core is not critical are shown in Figure 5.3-1. Pressure/temperature curves using plant specific data such as materials, fluences and stresses will be developed in accordance with Technical Specifications, see Subsection 5.3.1.5, prior to plant hydrostatic test. ~~An updated Pressure Temperature Limits Report (PTLR) is developed in accordance with the Technical Specifications.~~

Operating Limits During Heatup, Cooldown, and Core Operation

Figure 5.3-2 specifies representative limits applicable for normal reactor operation, including anticipated operational occurrences. Pressure/temperature curves using plant specific data such as materials, fluences and stresses will be developed in accordance with Technical Specifications, see Subsection 5.3.1.5, prior to plant start-up.

Reactor Vessel Annealing

In-place annealing of the reactor vessel, because of radiation embrittlement, is not necessary because the vessel is predicted to maintain an equivalent safety margin in accordance with the procedures of 10CFR50 Appendix G, Paragraph IV.A.

Predicted Shift in RT_{NDT} and Drop in Upper-Shelf Energy

For design purposes, the adjusted reference nil ductility temperature and drop in the USE for the ESBWR vessel is predicted in accordance with the requirements of Regulatory Guide 1.99.

The calculations are based on the limits specified in Table 5.3-1 on copper and nickel in the weld and forging material.

The fluence analysis was performed using the NRC accepted methodology documented in Reference 5.3-1. The estimated peak fluence for the vessel base material (1/4 T) and the weld above the TAF (at the inside of the RPV) are provided in Table 5.3-4.

As required by 10CFR50 Appendix H, a surveillance program will be conducted in accordance with the requirements of ASTM E-185. The surveillance program will