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Subject: **Response to Portion of NRC Request for Additional Information Letter Nos. 276 and 287 Related to the ESBWR Design Certification – Safety Analyses – RAI Numbers 15.3-35 through 15.3-37**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) responses to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter Nos. 276 and 287 (References 1 and 2). GEH responses to RAI Numbers 15.3-35 through 15.3-37 are addressed in Enclosure 1. DCD and Licensing Topical Report (LTR) markups associated with the response to RAI 15.3-35 are provided in Enclosure 2.

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

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

References:

1. MFN 08-957, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 276 Related To ESBWR Design Certification Application*, dated December 4, 2008
2. MFN 09-018, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 287 Related To ESBWR Design Certification Application*, dated January 7, 2009

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter Nos. 276 and 287 Related to ESBWR Design Certification Application – Safety Analyses – RAI Numbers 15.3-35 through 15.3-37
2. MFN 09-140 RAI 15.3-35 DCD and LTR Markups

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	eDRFs	0000-0096-8067R1

Enclosure 1

MFN 09-140

Response to Portion of NRC Request for Additional Information Letter Nos. 276 and 287 Related to ESBWR Design Certification Application

Safety Analyses

RAI Numbers 15.3-35 through 15.3-37

NRC RAI 15.3-35:

Per a teleconference with GEH on 11/13/2008 and follow- up review the staff requires clarifications for the following Chapter 15 figures (15.3-1a, 15.3-1g, 15.3-5g, and also 15.2-1g):

The clarifications identified below also pertain to Figure 15.2-1g and 15.3-5g.

(1) The significant MCPR differences on Fig. 15.3-1g "Loss of Feedwater Heating with SCRRI/SRI Failure" compared to Rev. 3 - The staff understands the use of the R-factors in the TRACG, PANAC11 and MONICORE etc codes. However, it seems that an adjustment of the numerical MCPR value is required to retain the MCPR value in Figure 15.3-1g to the same level as that in DCD Rev. 3, to be consistent with the actual MCPR at the initiation of the transient. This will be in synch with the statement that the R-factors have a negligible effect on the delta CPR/ICPR value for the transient while the MCPR itself is sensitive.

(2) The stable state shown on Fig. 15.3-1a at 117% rated power achieved at 300 seconds - GEH needs to modify Fig. 15.3-1a to indicate an acceptable end of the transient regarding power level. The statement that "no operator action is required" should be justified, clarified or eliminated.

(3) The apparent shift in the MCPR due to the specific R-factors in the TRACG analysis and

(4) The modified R-factors, seemingly, are used in this analysis and the inadvertent IC initiation transient analysis only.

GEH Response:

(1),(3),(4) The Anticipated Operational Occurrences (AOO) and Infrequent Event analyses documented in ESBWR DCD Rev 4, Chapter 15 were all performed using a channel R-factor of 1.04. In DCD Rev 5, the following cases were revised and the channel R-factor were also changed for each case:

15.2.1.1 Loss of Feedwater Heating

15.2.2.1 Fast Closure of One Turbine Control Valve

15.2.2.3 Generator Load Rejection With A Single Failure in the Turbine Bypass System

15.2.4.1 Inadvertent Isolation Condenser Initiation

15.3.1 Loss of Feedwater Heating With Failure of SCRRI and SRI

15.3.5 Generator Load Rejection with Total Turbine Bypass Failure

The difference in R-factor results in differences in the Initial Critical Power Ratio (CPR) (ICPR) and Minimum CPR (MCPR) values in comparison between DCD Rev 4 and DCD Rev 5 analyses. However, for AOO and Infrequent Event evaluations, the Initial CPR or MCPR independently are not as meaningful as (ICPR-MCPR)/ICPR

(ΔCPR/ICPR), the input used to determine the Operating Limit CPR (OLMCPR) (see Reference (1)). Because the ICPR is in both the numerator and denominator and because the R-factor effect on the ICPR and MCPR are very similar, there is no appreciable effect on the ΔCPR/ICPR. This is noted in Reference (2) where it states: "The ΔCPR/ICPR is independent relative to the uncertainties that affect the ICPR…"

Because the R-factor has no appreciable effect on the main parameter of importance, ΔCPR/ICPR, no DCD change is needed.

(2) As noted in the event description and in DCD Section 15.1, there is no operator action required to mitigate the Loss-of-Feedwater-Heating-with-SCRRI/SRI-Failure event, and results are acceptable with no operator action; however, operators would note that power is higher than rated conditions and would not allow power to remain elevated. The following sentence is added to both DCD Subsection 15.3.1.2 and NEDO-33337 Subsection 2.4.1.2, "However, operators will not permit reactor operation at elevated powers, and will lower power in accordance with the applicant's license and regulations."

References

(1) NEDE-33083P, Supplement 3, "TRACG Application for ESBWR Transient Analyses," December 2007.

(2) NEDE-32906-A, Revision 3, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," September 2006.

DCD or LTR Impact:

LTR NEDO-33337 Revision 1 will reflect the attached markup.

DCD Tier 2, Chapter 15, Revision 6 will reflect the attached markup.

<u>NRC RAI 15.3-36</u>:

DCD Rev 5, Sections 15.3.5 and 15.3.6. These sections describe almost identical transients resulting in a large power spike, followed by a fast scram, etc. With traces almost the same, the first shows a MCPR of 1.38 while the other has a MCPR of 1.23. Please discuss the results of 15.3.5 vs. those in 15.3.6 and justify the differences. Also, extend the calculation to at least 50 seconds to include the effect of the HP CRD injection, revise the response to RAI 15.3-12, and justify or revise the conclusion regarding fuel damage.

<u>GEH Response</u>:

1) The Load Rejection with Total Turbine Bypass Failure and the Turbine Trip with Total Turbine Bypass Failure are essentially the same event with the Load Rejection case slightly more limiting. There are two main differences between these two cases in DCD Rev 5:

a. <u>Different R-factor used for each case</u>

As noted in the response to RAI 15.3-35, the Generator Load Rejection with Total Turbine Bypass Failure event was updated in DCD Revision 5 with a different R-factor; whereas, the Turbine Trip with Total Turbine Bypass Failure event was not revised. Therefore, there is an apparent difference in the CPR values. As discussed in the response to RAI 15.3-35, this has no appreciable effect on the ΔCPR/ICPR.

b. <u>Turbine control valve stroke closing time vs. the turbine stop valve stroke closing time</u>

In the Load Rejection event, the turbine control valves (TCVs) shut in 0.08 seconds (see Table 15.3-6a), and in the Turbine Trip event, the turbine stop valves shut in 0.10 seconds (see Table 15.3-7). Because of the similarities, the Turbine Trip event follows the same trend as the Generator Load Rejection event. The Load Rejection case, which is more limiting, shows the long-term response (50 seconds). For a short-term response, the turbine trip case is only shown out to 10 seconds.

The Generator Load Rejection DCD figures shows the level recovery due to feedwater flow and also shows the % flow of the High Pressure Control Rod Drive (HP CRD). Because the Turbine Trip case is almost identical, the same trend applies; therefore, there is no need to extend the calculation to 50 seconds.

Because the events are nearly identical, no DCD or LTR update is needed.

2) “Revise the response to RAI 15.3-12, and justify or revise the conclusion regarding fuel damage”.

The response and conclusion to RAI 15.3-12 (MFN 07-038 dated January 30, 2009) is still applicable. Analyses show that the radiological consequences are less than 2.5 rem Total Effective Dose Equivalent (TEDE). DCD subsection 15.3.6.4 (Turbine Trip section) states: “The number of rods in boiling transition is confirmed to be under 1000 using the ΔCPR/ICPR results…" Subsection 15.3.6.5 states: "The off site dose for this event is less than 25 mSv (2.5 rem) Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures." Therefore, no change to the RAI 15.3-12 response is needed.

<u>DCD or LTR Impact</u>:

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

No changes to any LTR will be made in response to this RAI.

NRC RAI 15.3-37:

In DCD Rev. 5, Section 15.3.15, "Stuck Open Safety Relief Valve", the CRDs are credited to recover vessel water level. The IC's are also activated but their contribution does not appear until about 100 seconds into the transient. Because the CRDs are not safety-related, credit cannot be taken in the analyses.

A stuck open safety relief valve releases significant amounts of steam to the suppression pool lowering the water level fast and scramming the reactor automatically at L2. Because of the stuck open valve, the plant continues to lose water and the scenario in the DCD activates the CRDs that provide water quickly to recover level. Table 15.3-12 does not indicate when, but around 100 seconds into the transient the IC's are activated. Figure 15.3-9b indicates a surge of IC condensate around 100 seconds. Table 15.3-12 indicates that about 320 seconds the CRDs are deactivated because water level has been recovered. The recovery comes from the contribution of both the CRDs and the IC's. Figure 15.3-9c indicates a very fast rise in water level almost immediately after the scram. There is a long time between the L2 level (scram level) and the IC contribution to level increase. The level is reversed almost immediately with the CRDs.

Please provide an analysis of this event without crediting the CRDs to demonstrate that the core remains covered.

GEH Response:

Based on the context of the RAI, it is assumed that the activation of the High Pressure Control Rod Drive (HP CRD) makeup water is questioned, not credit of the HCU and CRDs which have safety-related functions. Water inserted in the core from the HCU/CRD to the RPV during scram can be credited, and is credited in the LOCA analysis (Reference (3)), but is not credited in the Stuck-Open-Safety-Relief-Valve calculation. Also, for clarification, the scram level is level 3, not level 2. Level 2 is the level that initiates the ICS and MSIV isolation after a 30-second delay as noted in DCD Tier 2, Table 15.2-23. Finally, it is noted that the feedwater system is assumed to function normally in this event and is the main reason for the initial increase in level.

The HP CRD makeup water (not safety-related) is included in the analysis, but is not the primary success path for this event. As shown in DCD Figure 15.1-30, the Automatic Depressurization System is initiated and the Gravity-Driven Cooling System is started to control water level in the event that the HP CRD is unavailable and water level reaches Level 1. The limiting event for low water level for an Infrequent Event (IE) is an Inadvertent Opening of a Depressurization Valve (DPV) with auxiliary power unavailable. This event is analyzed in Subsection 15.3.14. In the respective subsection it states: “If auxiliary power is not available, the sequence of events is similar to the Main Steam Line Break (MSLB) sequence given in Table 6.3-8.” The MSLB analysis demonstrates that the vessel inventory is maintained above the top of active fuel without the use of HP CRD flow. The description and analysis for the MSLB is presented in the subsection that corresponds to Table 6.3-8. The stuck open SRV event is bounded by this event with respect to core cooling and inventory control; therefore, there is no need to reanalyze this event.

Reference

(3) Response to RAI 6.3-87, MFN 08-802, “Response to Portion of NRC Request for Additional Information Letter No. 229 Related to ESBWR Design Certification Application - Emergency Core Cooling Systems RAI Numbers 6.3-83 and 6.3-87,” November 3, 2008.

DCD or LTR Impact:

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

No changes to any LTR will be made in response to this RAI.

Enclosure 2

MFN 09-140

RAI 15.3-35 DCD and LTR Markups

2.4 ANALYSIS OF INFREQUENT EVENTS

In Reference 2.4-3, ~~GE Nuclear Energy, "~~ESBWR D~~esign~~ C~~ontrol~~ D~~ocument" 26A6642BP, Revision 3, February 2007~~, ~~a~~Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A, and Infrequent Events. Section 15.0 of the DCD, Tier 2, Chapter 15 describes the licensing basis for this categorization.

The input parameters, initial conditions, and assumptions in Tables 2.3-1, 2 and 3 are applied in the TRACG calculations, based on the initial core in Reference 2.4-1 for the Infrequent Events addressed in Subsections 2.4.1 through 2.4.6 and Subsections 2.4.13 and 2.4.15. The summary of the Infrequent Events analyses ~~is~~ are given in Tables 2.4-1a and Table 2.4-1b.

The results of the system response analyses for the initial core design documented in Reference 2.4-1 are provided here. System response analyses bounding operation in the feedwater temperature operating domain is documented in Reference 2.4-2.

2.4.1 Loss of Feedwater Heating With Failure of SCRRI and SRI~~Selected Control Rod Run-In~~

2.4.1.1 Identification of Causes

Identification of causes is documented in the DCD, Tier 2, Chapter 15 (Reference 2.4-3).

2.4.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 2.4-2 lists the sequence of events for Figure 2.4-1. There is no operator action required to mitigate the event. However, operators will not permit reactor operation at elevated powers, and will lower power in accordance with the applicant's license and regulations.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The high STPT scram is the primary protection system trip in mitigating the effects of this event. However, credit was not taken for this scram to consider the possibility that, for a similar case with a somewhat lower loss of heating, the scram setpoint might not be reached, while the consequences would only be slightly less severe for this case than the event analyzed here.

2.4.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 2.3-1. FW enthalpy resulting in the maximum possible power allowed by the STPT setpoint is assumed.

A feedwater temperature controller failure to minimum temperature demand can also result in an inadvertent reduction in feedwater temperature. While operating at normal conditions, a failure in the feedwater controller with minimum temperature demand could result in the opening of the high-pressure feedwater heater bypass valves. The feedwater temperature reduction is bounded by 55.6°C (100°F).

While operating at higher feedwater temperatures within the feedwater temperature increase region of the feedwater temperature operating domain (Figure 4.4-1), the No. 7 feedwater heater steam heating valves are open. A feedwater temperature controller failure to minimum temperature demand results in closure of the No. 7 feedwater heater steam heating valves, and subsequent opening of the high-pressure feedwater heater bypass valves. The resulting decrease in feedwater temperature is potentially greater than 55.6°C (100°F). The frequency of a feedwater temperature controller failure to minimum temperature demand is evaluated in Subsection 15A.3.5.2. For this event, the loss of feedwater heating is detected by ATLM and DPS, and each sends a signal to initiate the SCRRI/SRI function. This mitigates the power increase caused by the reduction in feedwater temperature.

It is very unlikely that both feedwater temperature controller and SCRRI/SRI fail; however, if both failures occur, the reactor scrams on high simulated thermal power. In this event, the simulated thermal power setpoint is a function of the feedwater temperature as shown in Table 15.2-1. The rate of change limit on the simulated thermal power limit is shown in Table 15.2-1. Crediting the simulated thermal power scram ensures that the event remains bounded by the analysis of the loss of a feedwater heating with failure of SCRRI/SRI starting from the maximum allowable feedwater temperature operating point documented in Subsection 2.4.1 of Reference 15.3-7. Because the event is bounded by the loss of feedwater heating event described in Subsection 2.4.1 of Reference 15.3-7, no further discussion of the feedwater controller failure with minimum temperature demand is included in this subsection.

15.3.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-2 lists the sequence of events for Figure 15.3-1. There is no operator action required to mitigate the event. However, operators will not permit reactor operation at elevated powers, and will lower power in accordance with the applicant's license and regulations.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it is assumed that the plant instrumentation and controls, plant protection, and reactor protection systems function normally.

The STPT scram is the primary protection system trip in mitigating the effects of this event.

15.3.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.