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MFN 08-272

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Subject: Response to Portion of NRC Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-63 S01

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. DCD Markups related to this response are provided in Enclosure 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 06-167, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application*, June 1, 2006
2. E-Mail from Shawn Williams, U.S. Nuclear Regulatory Commission, to George Wadkins, GE Hitachi Nuclear Energy, dated May 22, 2007 (ADAMS Accession Number ML071430342)

Enclosures:

1. MFN 08-272 - Response to Portion of NRC Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-63 S01
2. MFN 08-272 - Response to Portion of NRC Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-63 S01 – DCD Markups

cc:	AE Cubbage	USNRC (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	GB Stramback	GEH/San Jose (with enclosures)
	RE Brown	GEH/Wilmington (with enclosures)
	eDRF	0000-0077-3384

Enclosure 1

MFN 08-272

Response to Portion of NRC Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application

Containment Systems

RAI Number 6.2-63 S01

NRC RAI 6.2-63 S01:

The information provided in this response is necessary to support the basis for a reasonable assurance finding. Thus, please update DCD Tier 2 to include information provided in response to RAI 6.2-63.

GEH Response:

The information requested by RAI 6.2-63 consisted of energy source data for the limiting Feedwater Line Break (FWLB) and Main Steam Line Break (MSLB) cases, including the energy removed by the Passive Containment Cooling System (PCCS). This information was presented in graphical form using three separate time formats (i.e., time zero to 72 hours, time zero to 100 seconds, and time zero to 2000 seconds) for each of the two limiting inside containment break conditions. Information was also provided related to sensible heat release to containment for time zero to 2000 seconds and time zero to 72 hours periods.

The above information was incorporated into the DCD Tier 2,Section 6.2, in Revision 2 as Figures 6.2-9c-series (FWLB) and 6.2-10c-series (MSLB) for the nominal case results, and Figures 6.2-13c-series (FWLB) and 6.2-14c-series (MSLB) for the bounding case results as requested by this RAI. The only change from the RAI response information was that the figures with time zero to 100 seconds period were revised to a time zero to 500 seconds period. These figures have been maintained and updated, as necessary, through the current revision (DCD Tier 2, Revision 4). The reactor pressure vessel (RPV) sensible heat information will be incorporated as a new DCD Tier 2, Table 6.2-12d, and the RPV heat slab temperature profiles will be incorporated into new DCD Tier 2 Figures 6.2-9e1 and 6.2-9e2 for a FWLB, and Figures 6.2-10e1 and 6.2-10e2 for a MSLB.

DCD Impact:

DCD Tier 2, Subsection 6.2.1.3 will be revised, and Table 6.2-12d and Figures 6.2-9e1, 6.2-9e2, 6.2-10e1, and 6.2-10e2 will be added, as shown in the attached markup.

Enclosure 2

MFN 08-272

Response to Portion of NRC Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application

Containment Systems

RAI Number 6.2-63 S01

DCD Markups

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

Relevant to mass and energy analyses, this subsection addresses or references to other DCD locations that address the applicable requirements of GDC 50 and 10 CFR Part 50, Appendix K, paragraph I.A discussed in SRP 6.2.1.3 R1. The plant meets the requirements of

- GDC 50, as it relates to the containment being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment design can withstand the calculated pressure and temperature conditions resulting from any loss-of-coolant accident; and
- 10 CFR 50, Appendix K, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

In meeting the requirements of GDC 50 the following criteria, which pertain to the mass and energy analyses, are used.

- Sources of Energy
  - The sources of stored and generated energy that are considered in analyses of LOCAs include reactor power, decay heat, stored energy in the core and stored energy in the reactor coolant system metal, including the reactor vessel (see Table 6.2-12d and Figures 6.2-9e1, 6.2-9e2, 6.2-10e1, and 6.2-10e2) and reactor vessel internals;
  - Calculations of the energy available for release from the above sources are done in general accordance with the requirements of 10 CFR 50, Appendix K, paragraph I.A. However, additional conservatism is included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA; and
  - The requirements of paragraph I.B in Appendix K, concerning the prediction of fuel cladding swelling and rupture are not considered, to maximize the energy available for release from the core to the containment.
- Break Size and Location
  - The choice of break locations and types is discussed in Subsection 6.2.1.1.3;
  - Of several breaks postulated on the basis stated above, the break selected as the reference case yields the highest containment pressure consistent with the criteria for establishing the break location and area; and
  - Containment design basis calculations are performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.
- Calculations

  Following the procedure, documented in Reference 6.2-1, calculations of the mass and energy release rates for a LOCA are performed in a manner that conservatively establishes the containment internal design pressure (that is, maximizes the post-accident containment pressure).

A spectrum of breaks was considered and analyzed using GEH-developed and approved computer codes described in Reference 6.2-1. The summary of this evaluation is discussed in Subsection 6.2.1.1.3.

Table 6.2-12d

RPV Sensible Heat Data

Item	Weight (kg)	C_p (J/kg-K)	TRACG Vessel Location
Bottom Head	122016	515	L1, R1, 2, 3, 4
Cylinder	823810	515	L2:L20, R4
Top Head	142473	515	L21, R1, 2, 3, 4

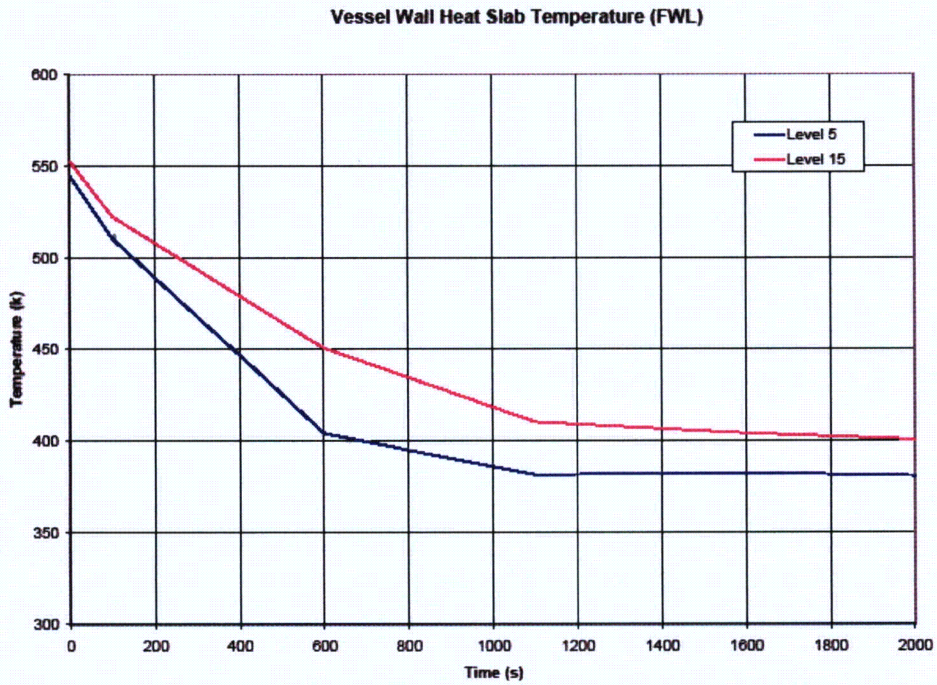

Figure 6.2-9e1. Feedwater Line Break (Nominal Case) - Reactor Pressure Vessel Heat Slab Temperatures (2000 S)

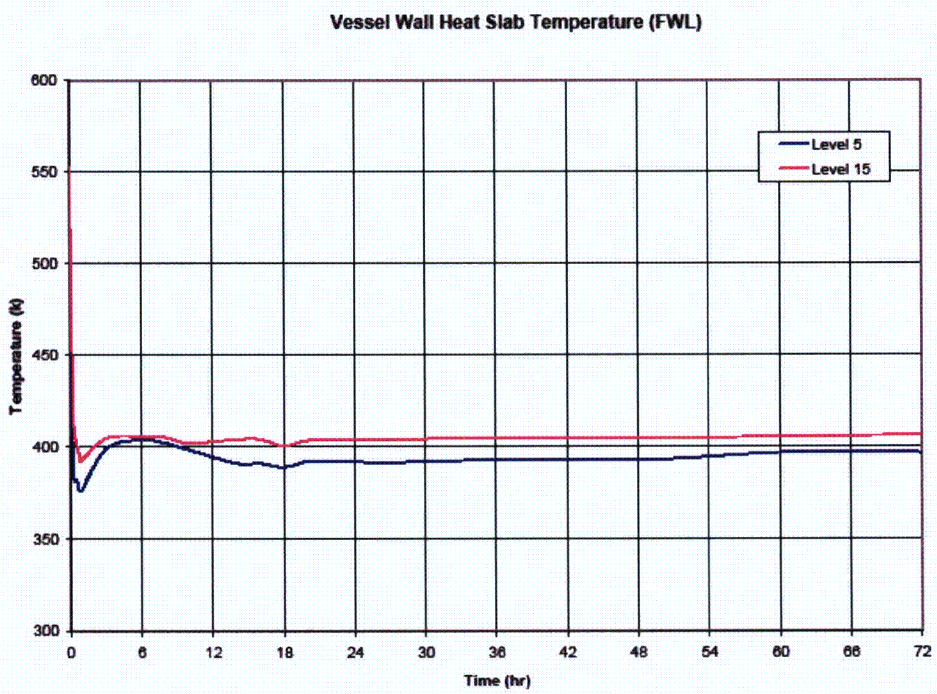

Figure 6.2-9e2. Feedwater Line Break (Nominal Case) - Reactor Pressure Vessel Heat Slab Temperatures (72 hrs)

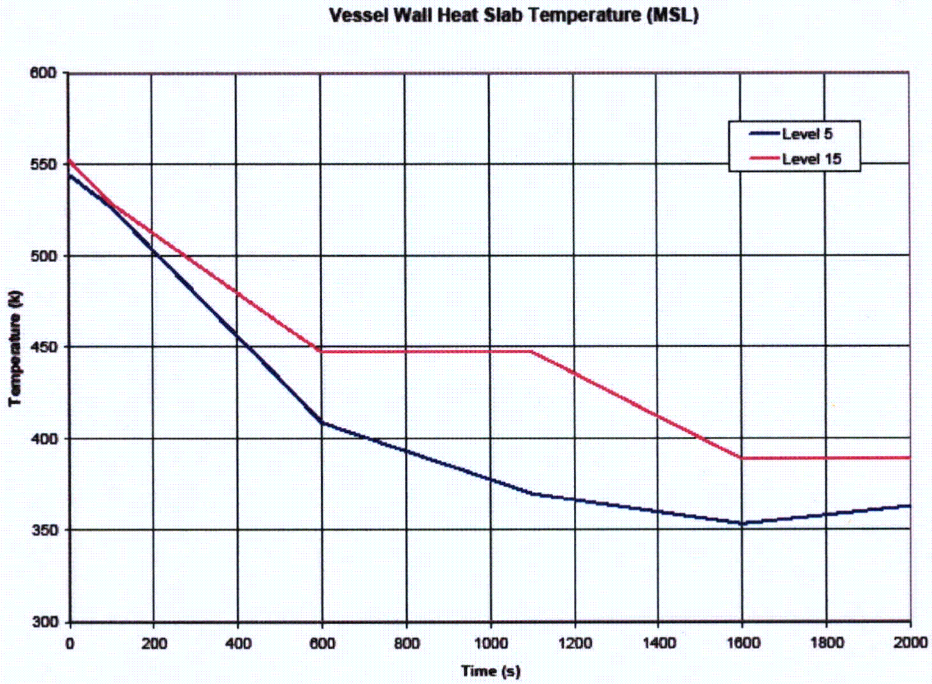

Figure 6.2-10e1. Main Steam Line Break (Nominal Case) - Reactor Pressure Vessel Heat Slab Temperatures (2000 S)

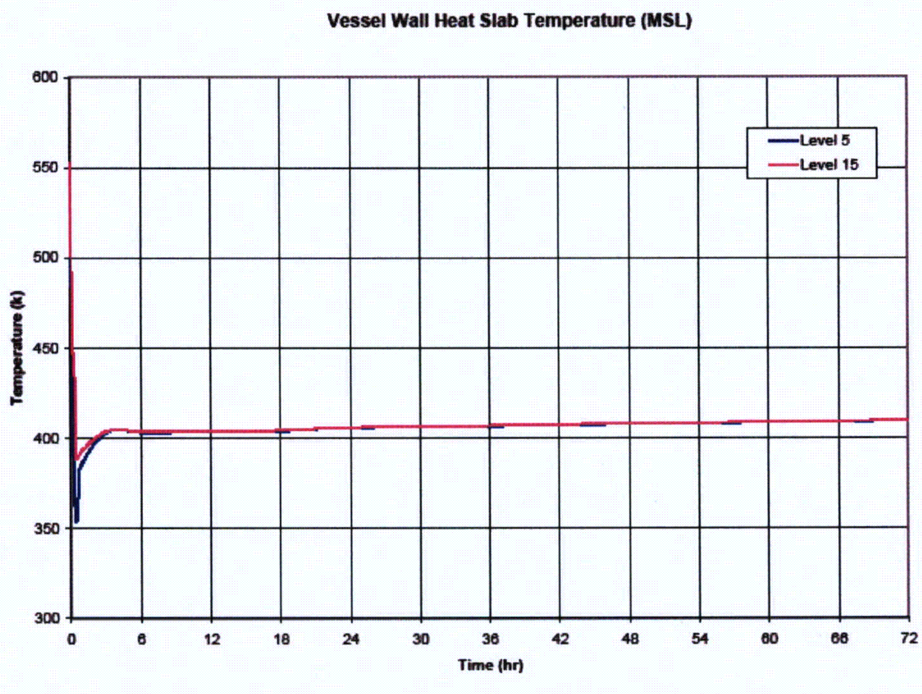

Figure 6.2-10e2. Main Steam Line Break (Nominal Case) - Reactor Pressure Vessel Heat Slab Temperatures (72 hrs)