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Proprietary Information Notice
This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon the removal of Enclosure 3.

MFN 08-419

Docket No. 52-010

May 1, 2008

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to Portion of NRC Request for Additional Information Letter No. 120 Related to ESBWR Design Certification Application - Emergency Core Cooling Systems - RAI Number 6.3-1 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.

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

As described in the enclosed RAI response, Enclosures 3 and 4 contain the GEH proprietary version of NEDC-33082P, ESBWR Scaling Report, Revision 2, and the non-proprietary version of NEDO-33082, ESBWR Scaling Report, Revision 2, respectively.

Enclosure 3 contains proprietary information as defined in 10 CFR 2.390. The affidavit contained in Enclosure 5 identifies that the information contained in Enclosure 3 has been handled and classified as proprietary to GEH. GEH hereby requests that the proprietary information in Enclosure 3 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. Enclosure 4 is the non-proprietary version of the report in Enclosure 3, which does not contain proprietary information and is suitable for public disclosure.

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NRO

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

References:

1. MFN 06-025, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 4 Related to ESBWR Design Certification Application*, January 11, 2006
2. MFN 07-717, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request for Additional Information Letter No. 120 Related to ESBWR Design Certification Application*, December 19, 2007

Enclosures:

1. MFN 08-419 - Response to Portion of NRC Request for Additional Information Letter No. 120 Related to ESBWR Design Certification Application - Emergency Core Cooling Systems - RAI Number 6.3-1 S01
2. MFN 08-419 - Response to Portion of NRC Request for Additional Information Letter No. 120 Related to ESBWR Design Certification Application - Emergency Core Cooling Systems - RAI Number 6.3-1 S01 - DCD Markups
3. NEDC-33082P, Revision 2 - ESBWR Scaling Report, April 2008 - GEH Proprietary Information
4. NEDO-33082, Revision 2 - ESBWR Scaling Report, April 2008 - Non-Proprietary Information
5. MFN 08-419 - Affidavit - James C. Kinsey - dated May 1, 2008

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	0000-0082-9695R1

Enclosure 1

MFN 08-419

Response to Portion of NRC Request for Additional Information Letter No. 120 Related to ESBWR Design Certification Application

Emergency Core Cooling Systems

RAI Number 6.3-1 S01

NRC RAI 6.3-1 S01:

Submit Revision 2 of NEDC-33082P, "ESBWR Scaling Report," to include the information in MFN 06-225, "Response to NRC Request for Additional Information Letter No. 4 Related to ESBWR Design Certification Application - ESBWR Scaling Analysis - RAI Number 6.3-1," July 18, 2006. Reference Rev. 2 of NEDC-33082P in the next revision of the DCD.

This request is necessary because staff relied on the additional information in MFN 06-225 to come to the conclusion that the scaling analysis is acceptable.

GEH Response:

Revision 2 of NEDC-33082P, "ESBWR Scaling Report," has been prepared to include the information in MFN 06-225, and is provided in Enclosure 3 of this response. The associated non-proprietary version, Revision 2 or NEDO-33082, "ESBWR Scaling Report," is provided in Enclosure 4 of this response.

Reference to NEDC-33082P, Revision 2, will be added to DCD Tier 2, Subsections 6.2.1.1.3 and 6.3.3, as new DCD Tier 2, References 6.2-9 and 6.3-4.

DCD Impact:

DCD Tier 2, Subsections 6.2.1.1.3 and 6.3.3 will be revised, and new DCD Tier 2, References 6.2-9 and 6.3-4 will be added, as shown in the attached markup.

Enclosure 2

MFN 08-419

Response to Portion of NRC Request for Additional Information Letter No. 120 Related to ESBWR Design Certification Application

Emergency Core Cooling Systems

RAI Number 6.3-1 S01

DCD Markups

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

vacuum, thereby drawing suppression pool water into the exhaust pipes. Vacuum relief valves are provided on the discharge piping to limit reflood water levels in the SRV discharge pipes, thus controlling the maximum SRV discharge bubble pressure resulting from a subsequent valve actuation and water clearing transient.

The WW design absolute pressure and design temperature are shown in Table 6.2-1. Table 6.2-2 shows the normal plant operating conditions for the allowed suppression pool water and WW airspace temperature.

After an accident, the nonsafety-related FAPCS may be available in the suppression pool cooling mode and/or containment spray mode to control the containment pressure and temperature conditions. Heat is removed via the FAPCS heat exchanger(s) to the Reactor Component Cooling Water System (RCCWS) and finally to the Plant Service Water System (PSWS). The FAPCS is described in Subsection 9.1.3.

There is sufficient water volume in the suppression pool to provide adequate submergence over the top of the upper row of horizontal vents, as well as the PCCS return vent, when water level in RPV reaches one meter above the top of active fuel and water is removed from the pool during post-LOCA equalization of pressure between RPV and the WW. Water inventory, including the GDCS, is sufficient to flood the RPV to at least one meter above the top of active fuel.

6.2.1.1.3 Design Evaluation

Summary Evaluation

The key design parameters for the containment and their calculated values under the DBA conditions are shown in Tables 6.2-1 and 6.2-5, respectively. Scaling analyses documented in Reference 6.2-9 show that the sub-scale integral test facilities, i. e., GIRAFFE/He and PANDA, adequately simulate the phenomena important to the post-LOCA long-term cooling of the ESBWR containment.

The evaluation of the containment design is based on the analyses of a postulated instantaneous guillotine rupture of a feedwater line, a main steam line, a GDCS injection line, and a bottom head drain line. For plant operation with nominal feedwater temperature, the analysis results are discussed in this subsection. For plant operation with feedwater temperature maneuvering (increase and reduction), the limiting breaks were evaluated and results are discussed in Reference 6.2-7. Specifically, the initial feedwater temperature was varied from 160°C (320°F) to 252.2°C (486°F) for the bounding Feedwater Line Break (FWLB) and Main Steam Line Break (MSLB) analyses with failure of one DPV. The calculations were run for 72 hours, and the maximum DW pressure remained below the containment design pressure of 45.3 psig with adequate margin, similar to those shown in Table 6.2-5. The variation of the maximum DW pressure was small with respect to the initial feedwater temperature.

Table 6.2-6 provides the nominal and bounding values for the plant initial and operating conditions for this evaluation. This evaluation utilizes the GE Hitachi Nuclear Energy (GEH) computer code TRACG (Reference 6.2-1). NRC has reviewed and approved the application of TRACG to ESBWR LOCA analyses, per the application methodology outlined in the report. The confirmatory items in the Staff's Safety Evaluation Report (SER) (Reference 6.2-1) concerning the TRACG computer code are addressed and provided in References 6.2-3 and 6.2-4. TRACG is applicable to LOCAs covering the complete spectrum of pipe break sizes,

6.2.9 References

6.2-1 GE Nuclear Energy, "TRACG Application for ESBWR," NEDC-33083P-A, Class III, (Proprietary), March 2005, and NEDO-33083-A, Class I (Non-proprietary), October 2005.

6.2-2 Galletly, G.D., "A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells under Internal Pressure," ASME Journal of Pressure Vessel Technology, Vol.108, November 1986.

6.2-3 GE letter from David H. Hinds to U.S. Regulatory Commission, TRACG LOCA SER Confirmatory Items (TAC # MC 8168), Enclosure 2, Reactor pressure Vessel (RPV) Level Response for the Long Term PCCS Period, Phenomena Identification and Ranking Table, and Major Design Changes from Pre-Application Review Design to DCD Design, MFN 05-105, October 6, 2005.

6.2-4 GE letter from David H. Hinds to U.S. Regulatory Commission, Revised Response – GE Response to Results of NRC Acceptance Review for ESBWR Design Certification Application – Item 2, MFN 06-094, March 28, 2006.

6.2-5 Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Vol. 87, P 134, February 1965.

6.2-6 (Deleted)

6.2-7 GE Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338, ~~scheduled September~~Class I, October 2007.

6.2-8 Moody, F.J. "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels," General Electric Company, Report No. NEDO-21052, September 1975.

6.2-9 GE Hitachi Nuclear Energy, "ESBWR Scaling Report," NEDC-33082P, Revision 2, Class III (Proprietary), April 2008; NEDO-33082, Revision 2, Class I (Non-proprietary), April 2008.

6.3.3 ECCS Performance Evaluation

Performance of the ECCS network is determined by evaluating the system response to an instantaneous break of a pipe. The analyses included in this subsection demonstrate the adequacy of ESBWR ECCS network performance for the entire spectrum of postulated break sizes. Scaling analyses documented in Reference 6.3-4 show that the sub-scale integral test facilities, i. e., GIST and GIRAFFE/SIT, adequately simulate the phenomena important to the ESBWR ECCS performance after a postulated pipe break.

The analyses are based upon the bundle design shown within Section 4.3 and were performed with the TRACG model. For plant operation with nominal feedwater temperature, the analysis results are discussed in Subsection 6.3.3.7. For plant operation with feedwater temperature maneuvering (increase and reduction), the limiting breaks were evaluated for the initial core and results are discussed in Reference 6.3-3. Specifically, the initial feedwater temperature was varied from 160°C (320°F) to 252.2°C (486°F) for both the nominal and bounding GDCS injection line break (GDLB) analyses with failure of one SRV or one GDCS injection valve or one DPV. For all cases, the reactor core remained covered with adequate margin with no cladding heat up, similar to that shown in Table 6.3-5. Also, the variation of minimum chimney static head level and the minimum downcomer collapsed water level was small with respect to the initial feedwater temperature. As explained in Section 1.2 of Reference 6.3-3, the nuclear characteristic of the core (initial vs. equilibrium) is not an important parameter during LOCA. Therefore, the results presented in Table 6.3-5 and Reference 6.3-3 are valid for all cycles.

The Chapter 15 accidents for which ECCS operation is required are:

- Feedwater Line Break;
- Spectrum of BWR Steam System Piping Failures Outside Containment; and
- Loss-of-Coolant Accidents (inside containment).

Chapter 15 provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Technical Specifications

The MLHGR operating limits, used in the ECCS performance analysis, are documented in each cycle-specific Core Operating Limits Report (COLR), which is referenced by the Technical Specifications. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed within Subsections 6.3.2 and 6.3.3.9. Limits on minimum suppression pool water level are discussed in Subsection 6.2.1.1.2 and Table 6.2-3.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10 CFR 50.46, are evaluated below.

Criterion 1: Peak Cladding Temperature (PCT)

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F," which is equivalent to 1204°C. Conformance to Criterion 1 is shown for the system response analyses within Subsection 6.3.3.7 and specifically in Table 6.3-5 (Summary of LOCA Analysis Results).

6.3.7 References

6.3-1 GE Nuclear Energy, "Depressurization Valve Development Test Program Final Report," GEFR-00879, October 1990.

6.3-2 GE Nuclear Energy, "TRACG Application for ESBWR," NEDC-33083P-A, Class III (Proprietary), March 2005 and NEDO-33083-A, Class I (Non-proprietary), October 2005.

6.3-3 GE Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," NEDO-33338, Class I, ~~scheduled~~ October 2007.

6.3-4 GE Hitachi Nuclear Energy, "ESBWR Scaling Report," NEDC-33082P, Revision 2, Class III (Proprietary), April 2008; NEDO-33082, Revision 2, Class I (Non-proprietary), April 2008.