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MFN 08-323

Docket No. 52-010

April 21, 2008

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

Subject: **Response to NRC Request for Additional Information Letter No. 157 Related to the ESBWR Design Certification – Safety Analyses – RAI Number 15.0-31**

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 dated March 5, 2008. GEH response to RAI Number 15.0-31 is addressed in Enclosure 1. DCD Markups related to this response is provided in Enclosure 2.

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

Sincerely,

for

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

References:

1. MFN 08-234, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 157 Related To ESBWR Design Certification Application*, dated March 5, 2008.

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 157 Related to ESBWR Design Certification Application – Safety Analyses – RAI Number 15.0-31
2. Response to Portion of NRC Request for Additional Information Letter No. 157 Related to ESBWR Design Certification Application – Safety Analyses – RAI Number 15.0-31 – DCD Markups

cc:	AE Cubbage	USNRC (with enclosure)
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	RE Brown	GEH/Wilmington (with enclosure)
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Enclosure 1

MFN 08-323

Response to Portion of NRC Request for Additional Information Letter No. 157 Related to ESBWR Design Certification Application

Safety Analyses

RAI Number 15.0-31

NRC RAI 15.0-31:

Submit a summary of the operation and analysis of the ESBWR Feedwater Temperature Operating Domain in Chapter 15 of the DCD as an Appendix. Grand Gulf Updated FSAR Section 15D.13, "Feedwater out of service in the maximum extended operating domain" may be used as an example of the scope and format of the submittal.

GEH Response:

GEH agrees with the NRC request and proposes the following additions and markups in several chapters and sections of the DCD since the ESBWR Power – Feedwater Temperature Operating Domain affects several chapters including Chapter 15.

Chapter 4 additions:

a) The ESBWR Power – Feedwater Temperature Operating Domain is added as Figure 4.4-1 with reference from Subsection 4.4.4.3.

b) A new Subsection 4D.1.6 (Stability Performance for Initial Core and Feedwater Temperature Variation) is added to summarize the stability performance with respect to the feedwater temperature variation.

Chapter 6 additions:

a) Subsection 6.2.1.1.3 (Summary Evaluation) is enhanced to incorporate the effect of feedwater temperature variation on containment pressure due to Feedwater Line Break (FWLB) and Main Steam Like Break (MSLB).

b) Subsection 6.3.3 (ECCS Performance Evaluation) is enhanced to incorporate the effect of feedwater temperature variation on emergency core cooling system (ECCS) performance due to gravity drain cooling system (GDCS) injection line break (GDLB).

Chapter 15 additions:

a) A new Appendix 15D (EFFECT OF FEEDWATER TEMPERATURE VARIATION) is added to summarize the results of anticipated operational occurrences (AOOs), Infrequent Events, Special Events and other analyses with respect to the feedwater temperature variation.

b) This new Appendix 15D is referenced from the appropriate subsections of Sections 15.2, 15.3 and 15.5.

DCD Impact:

DCD Tier 2, Chapters 4, 6 and 15 will be revised as noted on the attached markups.

Enclosure 2

MFN 08-323

Response to Portion of NRC Request for Additional Information Letter No. 157 Related to ESBWR Design Certification Application

Safety Analyses

RAI Number 15.0-31

DCD Markups

4.4.4.3 Power/Flow Operating Map

The core power-flow map is only a single line and there is no active control of the core flow at a given power level. To provide additional operational flexibility, a core power – feedwater temperature operating map or domain is developed. The system hardware and control system required to develop and implement such an operating ~~map~~ domain is discussed in Subsections 10.4.7.2.2.3 and 7.7.3, respectively. The typical operating domain of the ESBWR in the core power-feedwater temperature map is shown in Figure 4.4-1. The details are discussed in Reference 4.4-18.

4.4.4.4 Temperature-Power Operating Map

The temperature-power operating map, typically employed for Pressurized Water Reactor (PWR), is not applicable to the ESBWR. However, as mentioned in Subsection 4.4.4.3 above, a power-feedwater temperature operating map is developed for the ESBWR.

4.4.4.5 Load Following Characteristics

Load following is implemented through the Plant Automation System (PAS). This is described in Chapter 7 Subsection 7.7.4.

4.4.4.6 Thermal-Hydraulic Characteristics Summary Tables

The thermal-hydraulic characteristics are provided in Table 4.4-1a, Table 4.4-1b and Table 4.4-6. The axial power distributions for the average and hot channels are shown in Table 4.4-4a and Table 4.4-4b. The axial distribution of void fractions for the average power channel and the hot channel are given in Table 4.4-2a and Table 4.4-2b. The core average and core maximum exit void fractions are also provided. Similar distributions for coolant flow quality are provided in Table 4.4-3a and Table 4.4-3b.

4.4.4.7 Inadequate Core Cooling (ICC) Monitoring System

The issue of an Inadequate Core Cooling (ICC) monitoring system is discussed in Appendix 1A (Response to TMI Related Matters), specifically, TMI Item II.F.2 in Table 1A-1 (TMI Action Plan Items) addresses this issue related to the ESBWR.

4.4.5 Loose-Parts Monitoring System

The Loose Parts Monitoring System (LPMS) has been withdrawn from the ESBWR design certification; therefore Regulatory Guide 1.133 is no longer applicable to the ESBWR design. This decision is supported by information documented in the License Topical Report, "Regulatory Relaxation for BWR Loose Parts Monitoring Systems", written by the BWR Owner's Group (Reference 4.4-19).

The ESBWR design and operation minimizes the potential for loose parts in the reactor pressure vessel. The ESBWR design takes into consideration material selection for critical components, and utilizes Flow Induced Vibration (FIV) testing and temporary strainers during startup to prevent loose parts from entering the reactor vessel. Foreign Materials Exclusion (FME) programs and underwater vessel inspections are employed to prevent loose parts from entering the reactor vessel. The ESBWR is capable of performing its safety-related functions without the LPMS.

4.4.6 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and remain within required limits throughout core lifetime, are discussed in Chapter 14.

4.4.7 COL Information

None.

4.4.7.1 (Deleted)

4.4.8 References

4.4-1 General Electric Company, "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299A, October 1976.

4.4-2 General Electric Company, "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A, August 1976.

4.4-3 General Electric Company, "Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations," NEDO-21215, March 1976.

4.4-4 R.C. Martinelli and D.E. Nelson, "Prediction of Pressure Drops During Forced Convection Boiling of Water," ASME Trans., 70, 695-702, 1948.

4.4-5 C.J. Baroczy, "A Systematic Correlation for Two-Phase Pressure Drop," Heat Transfer Conference (Los Angeles), AIChE, reprint No. 37, 1965.

4.4-6 N. Zuber and J. A. Findlay, "Average Volumetric Concentration in Two-Phase Flow Systems," Transactions of the ASME Journal of Heat Transfer, November 1965.

4.4-7 W. H. Jens and P. A. Lottes, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High Pressure Water," USAEC Report ANL-4627, 1951.

4.4-8 General Electric Company, "General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application," NEDO-10958-A, January 1977.

4.4-9 GE Nuclear Energy, "TRACG Application for ESBWR," NEDE-33083P-A Revision 0, Class III (Proprietary), March 2005.

4.4-10 GE Hitachi Nuclear Energy, "Licensing Topical Report TRACG Model Description", NEDE-32176P, Revision ~~3~~4, Class III (Proprietary), ~~April 2006~~January 2008, NEDO-32176, Revision 4, Class I (Non-proprietary), January 2008.

4.4-11 GE Nuclear Energy, "Licensing Topical Report TRACG Qualification," NEDE-32177P Revision ~~2~~3, Class III (Proprietary), ~~January 2000~~August 2007.

4.4-12 [~~GE~~ *Global Nuclear ~~Energy~~Fuel, "GE14 for ESBWR-Critical Power Correlation, Uncertainty, and OLMCPR Development", NEDC-33237P, Revision ~~2~~3, Class III (Proprietary), ~~April~~ December 2007.*]*

4.4-13 GE Nuclear Energy, "Methodology and Uncertainties for Safety Limit MCPR Evaluations", NEDC–32601P-A, Class III (Proprietary), August 1999.

4.4-14 GE Nuclear Energy, "GE14 Pressure Drop Characteristics", NEDC-33238P, Class III (Proprietary), December 2005.

4.4-15 "TASC-03A, A Computer Program for Transient Analysis of a Single Channel", NEDC-32084P-A, Revision 2, Class III (Proprietary), July 2002.

4.4-16 Letter, J.S. Charnley (GE) to C. O. Thomas (NRC), Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A, January 25, 1986.

4.4-17 GE Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analysis," NEDO-33337, ~~scheduled September~~Class I, October 2007.

4.4-18 GE Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," NEDO-33338, ~~scheduled September~~Class I, October 2007.

4.4-19 GE Nuclear Energy, "Regulatory Relaxation for BWR Loose Parts Monitoring Systems," BWR Owner's Group Licensing Topical Report NEDC-32975P-A, Class III (Proprietary), Revision 0, February 2001.

4.4-20 Global Nuclear Fuel, "GE14E for ESBWR Initial Core Nuclear Design Report," NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

4.4-21 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report," NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non proprietary), Revision 2, April 2007.

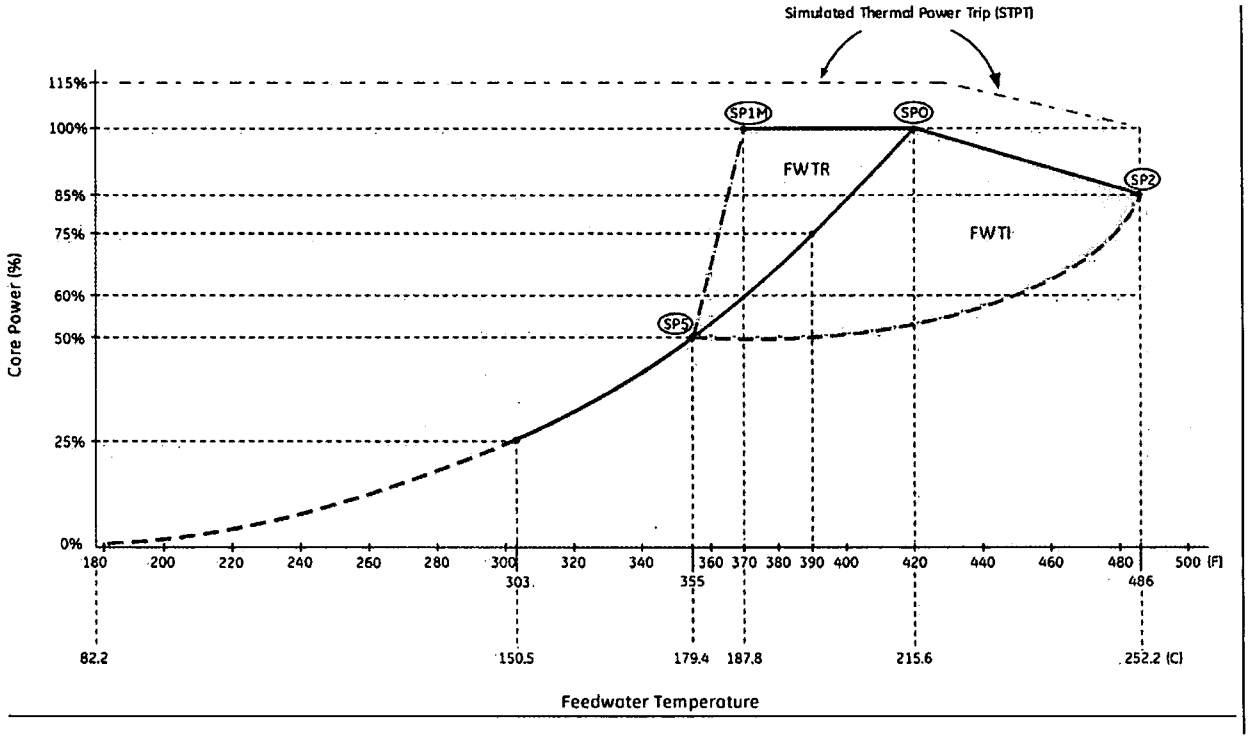

Figure 4.4-1. Typical ESBWR Power – Feedwater Temperature Operating Domain

4D.1.6 Stability Performance for Initial Core and Feedwater Temperature Variation

The demonstration of stability margins, in the previous subsections, is performed for an equilibrium core based on the GE14 fuel design described in Section 4.2. Similar studies are performed with an initial core and with variation of feedwater temperature from 187.8°C (370°F) to 252.2°C (486°F). The decay ratios for the limiting regional stability mode at baseline cases and during limiting AOO are within the design limits. The details are presented in References 4D-20 and 4D-21.

If a different core design is chosen, verification will be required to confirm that the design limits are met. If the baseline stability decay ratios are higher than the calculated values as presented in Table 4D-2, the statistical analysis of decay ratios will be performed and the results checked versus the design limits or acceptance criteria.

~~4D.1.6~~ 4D.1.7 Stability Performance During Anticipated Transients Without Scram

The ATWS mitigation design for the ESBWR is summarized in Subsection 15.5.4. This includes automatic feedwater runback and automatic boron injection. The TRACG analysis results presented in Subsection 15.5.4 confirm the conclusion that there are no stability issues during the ATWS transient.

4D.2 Stability Performance During Plant Startup

In contrast to operating BWRs, the ESBWR plant starts up without recirculation pumps. At low pressure, the initiation of voiding in the core and chimney causes perceptible changes in the driving head because of the large difference between liquid and vapor densities. Consequently, startup procedures are developed to assure smooth ascension in pressure and power.

Tests in experimental natural circulation loops (References 4D-12, 4D-13 and 4D-14) have identified two mechanisms for potential flow oscillations at low pressure. First, at very low flows, a periodic "geysering" flow oscillation was found to occur due to condensation of core exit vapor in the subcooled chimney region. Condensation–induced oscillations may occur under these conditions. The chimney subcooling and the rate of vapor production in the core determine the condensation rate. Oscillations of this kind are unlikely given the ESBWR startup procedures, which are designed to avoid vapor generation in the core prior to reaching saturated conditions in the chimney, and are similar to those of the natural circulation Dodewaard reactor. Dodewaard experienced no "geysering" oscillation in its 22 refuel cycles of operation. Second, initiation of vapor production in the chimney region leads to a reduction in hydrostatic head in the chimney and a resultant core flow increase. This, in turn, could cause voids to collapse in the chimney, leading to a reduction in flow. This type of oscillations termed 'Type 1' instability (Reference 4D-15) is unavoidable in a natural circulation reactor as the instability region has to be crossed prior to establishing a steady two phase voided region in the chimney, see Figure 4D-6. However, the magnitude of the flow oscillations is typically very small and this phenomenon had also never been observed at Dodewaard. In the final cycle of its operation, a special startup test was performed to probe the low-pressure portion of the startup trajectory. Though no oscillations were detectable on the APRMs, it was possible to infer the presence of small oscillations in core velocity from the auto correlation function of the Average Power Range Monitor (APRM) signal, see Figure 4D-7 and Reference 4D-16. These were small oscillations superposed on the core velocity with little, if any, reactivity impact, as the core flow

4D.4 References

4D-1 GE Nuclear Energy, J. S. Post and A. K. Chung, "ODYSY Application for Stability Licensing Calculations," NEDC-32992P-A, July 2001.

4D-2 GE Nuclear Energy, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," NEDO-31960-A, November 1995.

4D-3 GE Nuclear Energy, B. S. Shiralkar and Y. K. Cheung, "TRACG Application to ESBWR," NEDC-33083P-A, March 2005.

4D-4 GE Nuclear Energy, J. G. M. Andersen, et al., "TRACG Application for Anticipated Operational Occurrences Transient Analysis," NEDE-32906P-A, Revision ~~1~~3, ~~April~~September ~~2003~~2006.

4D-5 GE Nuclear Energy, F. T. Bolger and M. A. Holmes, "TRACG Application for ATWS Overpressure Transient Analysis," NEDE-32906P, Supplement 1-A, November 2003.

4D-6 GE Hitachi Nuclear Energy, J. G. M. Andersen, et al., "TRACG Model Description," NEDE-32176P, Revision ~~2~~4, Class III (Proprietary) January 2008, and NEDO-32176, Revision 4, Class I (Non-proprietary), January 2008~~December 1999~~.

4D-7 GE Nuclear Energy, J. G. M. Andersen, et al., "TRACG Qualification," NEDE-32177P, Revision ~~2~~3, ~~January~~ August 2007~~0~~.

4D-8 GE Nuclear Energy, J. R. Fitch, et al., "TRACG Qualification for SBWR," NEDC-32725P, Revision 1, Vol.1 and 2, August 2002.

4D-9 GE Nuclear Energy, J. R. Fitch, et al., "TRACG Qualification for ESBWR," NEDC-33080P, Revision 1, May 2005.

4D-10 USNRC, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," NUREG/CR-5249, December 1989.

4D-11 GE Nuclear Energy, B.S.Shiralkar, et al, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P-A, Supplement 1, Revision 1, ~~December~~ January 2008~~4~~.

4D-12 M. Aritomi, J. H. Chiang. M. Mori, "Fundamental Studies on Safety-Related Thermal-Hydraulics of Natural Circulation Boiling Parallel Channel Flow System under Startup Conditions (Mechanism of Geysering in Parallel Channels)," Nuclear Safety, Vol. 33, No.2, pp. 170-182, 1992.

4D-13 F. Inada, Y. Yasuo, "The Boiling Flow Instability of a Natural Circulation BWR with a Chimney at Low Pressure Startup," Proc. International Conference on the Design and Safety of Advanced Nuclear Power Plants (ANP '92), Tokyo, Japan, Paper 25.3, October 25-29, 1992.

4D-14 S. Kuran, M. Ishii, X. Sun, L. Cheng, Y. Xu, H. Yoon, S.T. Revankar, "Nuclear Coupled Flow Instability Study for Natural Circulation BWR Startup Transient," Paper N6P002, 6th International Conference on Nuclear Thermal Hydraulics, Operation and Safety (NUTHOS-6), Nara, Japan, October 2004.

4D-15 D.D.B. van Bragt and T.H.J.J. van der Hagen, "Stability of Natural Circulation Boiling Water Reactors: Part II - Parametric Study Of Coupled Neutronic-Thermohydraulic Stability," Nuclear Technology 121 (1998), 52-62.

4D-16 T.H.J.J. van der Hagen, F.J. van der Kaa, J. Karuza, W.H.M. Nissen, A.J.C. Stekelenburg, J.A.A. Wouters, "Startup of the Dodewaard Natural Circulation Boiling Water Reactor," GKN Report 92-017/FY/R, 1992.

4D-17 A. Manera and T.H.J.J. van der Hagen, "Stability Of Natural-Circulation-Cooled Boiling Water Reactors During Start-Up: Experimental Results," Nuclear Technology, 143 (2003), 77-88.

4D-18 Letter from K. Sedney, GE to A. Cubbage, USNRC, "GE Responses to NRC RAIs on NEDE-33083P, Supplement 1, TRACG Application for ESBWR Stability Analysis," MFN 05-52, June 2, 2005.

4D-19 GE Hitachi Nuclear Energy, "Man Machine Interface System (MMIS) and Human Factors Engineering (HFE) Implementation Plan," NEDO-33217, July 24, 2007.

4D-20 GE Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses," NEDO-33337, ~~scheduled September 2007~~Class I, October 2007.

4D-21 GE Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," NEDO-33338, Class I, October 2007.

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 is 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 are run for 72 hours, and the maximum DW pressure remains below the design pressure as shown in Table 6.2-1 with adequate margin, similar to those shown in Table 6.2-5. The variation of the maximum DW pressure is 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,

Fracture prevention of the containment pressure boundary is assured. The ESBWR meets the relevant requirements of the following regulations:

- General Design Criterion 1 (as it relates to the quality standards for design and fabrication) - See Subsection 3.1.1.1.
- General Design Criterion 16 (as it relates to the prevention of the release of radioactivity to the environment) - See Subsection 3.1.2.7.
- General Design Criterion 51 (as it relates to the reactor containment pressure boundary design) - See Subsection 3.1.5.2.

To meet the requirements of GDC 1, 16 and 51, the ferritic containment pressure boundary materials meet the fracture toughness criteria for ASME Section III Class 2 components. These criteria provide for a uniform review, consistent with the safety function of the containment pressure boundary within the context of Regulatory Guide 1.26, which assigns correspondence of Group B Quality Standard to ASME Code Section III Class 2.

6.2.8 COL Information

6.2-1-H ***Pipe Length from Containment to Inboard/Outboard Isolation Valve***

The COL Holder shall provide the missing information indicated in Tables 6.2-16 through 6.2-42. (Subsection 6.2.4.2.)

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.

6.3.3 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 is 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 remains 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 is 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.

acceptance criterion for AOOs because increases in the heat flux are not experienced. Consequently, this event does not need to be reanalyzed for specific core configurations.

15.2.5.3.3 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel or containment. Therefore, these barriers maintain their integrity and function as designed.

15.2.5.3.4 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.6 AOO Analysis Summary

The results of the system response analyses are presented in Table 15.2-5. Based on these results, the limiting AOO events have been identified. The potentially limiting events that establish the CPR operating limit are identified below. The results of the system response analyses for the initial core loading documented in Reference 15.2-3 are provided in Reference 15.2-4. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.2-5. A summary is provided in Appendix 15D.

For the core loading in Reference 15.2-2, the resulting OLMCPR is 1.30, using the methodologies listed in Subsections 4.4.3.1.3 and 4.4.2.1.3 and Reference 15.2-1. The operating limit for each fuel cycle is documented in the COLR in accordance with Technical Specifications. The following AOOs are potentially limiting with respect to OLMCPR:

- Loss of Feedwater Heating
- Closure of One Turbine Control Valve
- Generator Load Rejection with Turbine Bypass
- Generator Load Rejection with a Single Failure in the Turbine Bypass System
- Turbine Trip with Turbine Bypass
- Inadvertent Isolation Condenser Initiation

For the core loading in Reference 15.2-2, no adjustment to the Maximum Linear Heat Generation Rate (MLHGR) limits is needed to ensure compliance with the fuel thermal-mechanical acceptance criteria (cladding strain and fuel melt). The slow events, (fuel temperature follows fuel power in a quasi steady state condition) from the set of potentially limiting events, are evaluated for each fuel cycle using methodology consistent with Subsection 4.2.3.8 and Reference 15.2-1. Any required adjustment to the MLHGR limits is documented in the COLR in accordance with Technical Specification. For fast events, sufficient margin to fuel thermal-mechanical acceptance criteria exist. No fuel cycle specific evaluation is required.

15.2.7 COL Information

15.2-1-A (Deleted)

15.2-2-H (Deleted)

15.2-3-A (Deleted)

15.2.8 References

15.2-1 GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences Transient Analysis" NEDE-32906P-A, Revision ~~1~~3, ~~April~~ September 2006~~2003~~.

15.2-2 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report", NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non-proprietary), Revision 2, April 2007.

15.2-3 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.2-4 GE-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337, Class I, Revision 0, ~~Scheduled September 2007~~October 2007.

15.2-5 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338, Class I, Revision 0, ~~Scheduled September 2007~~October 2007.

15.3 ANALYSIS OF INFREQUENT EVENTS

Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A, or Infrequent Events. Section 15.0 describes the licensing basis for this categorization.

The input parameters, initial conditions, and assumptions in Tables 15.2-1, 2 and 3 are applied in the TRACG calculations, based on the equilibrium core in Reference 15.3-4, for the Infrequent Events addressed in Subsections 15.3.1 through 15.3.6 and Subsections 15.3.13 and 15.3.15. The summary of the Infrequent Events analyses is given in Tables 15.3-1a and 15.3-1b.

The results of the system response analyses for the initial core design documented in Reference 15.3-5 are provided in Reference 15.3-6. System response analyses bounding operation in the feedwater temperature operating domain is documented in Reference 15.3-7. A summary is provided in Appendix 15D.

15.3.1 Loss Of Feedwater Heating With Failure of Selected Control Rod Run-In

15.3.1.1 Identification of Causes

The loss of a feedwater (FW) heater can occur in at least two ways:

- Steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual FW cooling. In the second case, the FW bypasses the heater and no FW heating occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating.

This event conservatively assumes the loss of FW heating as shown on Table 15.2-1, causing an increase in core inlet subcooling and core power due to the negative void reactivity coefficient. However, the power increase is slow.

A loss of feedwater heating that results in a significant decrease in feedwater temperature is independently detected by the ATLMs and by the Diverse Protection System (DPS), either of which mitigates the event by initiating Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) functions, and scram is avoided.~~The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic provided to mitigate the effects of a loss of FW heating capability. The FWCS is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Rod Control and Information System (RC&IS) to initiate the Selected Control Rod Run-In and Select Rod Insertion (SCRRI/SRI) function to automatically reduce the reactor power and avoid a scram.~~ However, for this event, SCRRI/SRI is assumed to fail and reactor scram on high simulated thermal power is not credited due to uncertainties. Therefore a new steady state is reached.

15.3-2-H (Deleted)

15.3-3-A (Deleted)

15.3-4-A (Deleted)

15.3-5-A (Deleted)

15.3-6-A (Deleted)

15.3.18 References

15.3-1 GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel--United States Supplement," NEDE-24011-P-A-US, (Latest approved revision).

15.3-2 Deleted

15.3-3 FLN–2004–026, "GESTAR I Amendment 28 Revision 1, Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7," Margaret E. Harding to Mel B. Fields August 23, 2004

15.3-4 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report", NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non-proprietary), Revision 2, April 2007.

15.3-5 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.3-6 GE_-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337, Class I, Revision 0, ~~Scheduled September~~October 2007.

15.3-7 GE_-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338, Class I, Revision 0, ~~Scheduled September~~ October 2007.

15.3-8 General Electric Co., "Steady State Nuclear Methods," NEDE-30130-P-A, April 1985.

15.3-9 R. J. Reda (GE) to R. C. Jones, Jr. (NRC), "Implementation of Improved Steady-State Nuclear Methods," MFN-098-96, July 2, 1996.

15.3-10 G. A. Watford (GE) to E. D. Kendrick (NRC), "Implementation of Improved Steady-State Nuclear Methods," MFN-003-98, January 8, 1998.

15.3-11 NRC "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report GESTAR II Amendment 28, 'Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7,' Global Nuclear Fuel Project 712."

15.3-12 Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1.

15.3-13 Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0.

15.5 SPECIAL EVENT EVALUATIONS

15.5.1 Overpressure Protection

~~Results of the overpressure protection evaluation are provided in Subsection 5.2.2.~~

15.5.1.1 Method of Analysis

The method of analysis is approved by the United States Nuclear Regulatory Commission (NRC) or developed using criteria approved by the NRC. The acceptance criteria for overpressure protection are provided in Subsection 15.0.3.4.1. This analysis is required to demonstrate prevention of reactor coolant pressure boundary ASME Code Service Level B pressure limit(s).

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of three sources: (1) a direct valve position signal, (2) a flux signal, or (3) a high vessel pressure signal. The direct scram trip signal is derived from position switches mounted on the MSIVs. The pressure signal is derived from pressure transmitters piped to the vessel steam space.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

15.5.1.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions

Operating Conditions

- Operating power = 4590 MWt (102 % of nuclear boiler rated power);
- Absolute vessel dome pressure ≤ 7.17 MPa (1040 psia); and
- Steam flow = 2433 kg/s (19.31 Mlbm/hr).

These rated power conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe.

Pressurization Events

The overpressure protection system is capable of accommodating the most severe pressurization event. The ESBWR pressurization is mild relative to previous other BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization. The scram and initial pressurization drops the water level below the feedwater sparger; when the feedwater system performs as expected, the spray of subcooled water condenses steam in the vessel steam space and immediately terminates the pressurization. For purposes of overpressure protection analyses, the feedwater system is assumed to trip at the initiation of the event. The analyses of increase-in-reactor-pressure events are evaluated Subsection 15.2.2, where the

performance of the ICS is credited to prevent a lift of the SRVs or SVs. In order to evaluate the overpressure protection capability of the SRVs, no credit is taken in this evaluation for the ICS.

No credit is taken for the first scram signal that would occur (e.g., valve position for MSIV isolation). This is in accordance with NUREG-0800, Subsection 5.2.2, which requires that the reactor scram be initiated by the second safety-related signal from the Reactor Protection System (neutron flux for MSIV isolation, turbine trip and load rejection).

The evaluation of event behavior, based on the equilibrium core in Reference 15.5-6, demonstrates that MSIV closure, with scram occurring on high flux, (i.e., MSIV Closure With Flux Scram special event, MSIVF) is the most severe pressurization AOO event, the result for this event is similar to the Turbine Trip With Total Turbine Bypass Failure event evaluated in Subsection 15.3.6. Other fuel designs and core loading patterns, including loading patterns similar to Reference 15.5-6, do not affect the conclusions of this evaluation. Table 15.5-1a lists the systems that could initiate during a MSIV Closure With Flux Scram special event

The results of the overpressure protection analysis for the initial core loading documented in Reference 15.5-3 are provided in Reference 15.5-4. Overpressure protection analysis bounding operation in the feedwater temperature operating domain are documented in Reference 15.5-5. A summary is provided in Appendix 15D.

Evaluation Method

The evaluation method for overpressure protection events is the TRACG computer code as described in Reference 15.5-7.

SRV & Pressurization Event Analysis Specification

- Simulated valve group:
  - Spring-action safety mode – 1 valve credited in analysis
- Opening pressure setpoint (maximum safety limit):
  - Spring-action safety mode – Low Setpoint, Table 15.2-1
- Reclosure pressure setpoint (% of opening setpoint) both modes:
  - Maximum safety limit (used in analysis) 96
  - Minimum operational limit 90

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Conservative SRV response characteristics (Table 15.2-1) are also assumed; therefore, the analysis conservatively bounds all SRV operating conditions.

The RPS high flux scram settings assumed are provided in Table 15.2-1.

The MSIV design closure time range and the worst case (bounding) closure time assumed in this analysis are provided in Table 15.2-1.

15.5.1.3 Evaluation of Results

Total SRV Capacity

high, and the transient response to these perturbations is evaluated. The limiting case, where a perturbation was introduced at 25 seconds, is illustrated in Figure 15.5-9. The perturbations in the power are quickly damped out, indicating that the ESBWR operation remains stable during these events.

15.5.4.4 Conclusion

Based upon the results of this analysis, the proposed ATWS design for the ESBWR is satisfactory in mitigating the consequences of an ATWS. All performance requirements specified in Subsection 15.5.4.3.2 are met. It is also demonstrated that the plant operation remains stable during an ATWS event. The results of the system response analyses for the initial core loading documented in Reference 15.5-3 are provided in Reference 15.5-4. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.5-5. A summary is provided in Appendix 15D.

It is also concluded from results of the above analysis that automatic boron injection could mitigate the most limiting ATWS event with margin. Therefore, an automatic SLCS injection as a backup for ATWS mitigation is acceptable.

15.5.5 Station Blackout

The performance evaluation for Station Blackout (SBO) show conformance to the requirements of 10 CFR 50.63 and is presented in this subsection.

15.5.5.1 Acceptance Criteria

The design meets the following acceptance criteria:

- **Reactor Vessel Coolant Integrity** - Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- **Hot Shutdown Condition** - Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- **Containment Integrity** - If containment isolation is involved, the maximum containment and suppression pool pressures and temperatures are maintained below their design limits.

15.5.5.2 Analysis Assumptions

The analysis assumptions and inputs are summarized below.

- Reactor is operating initially at 102% of rated power/100% rated nominal core flow, nominal dome pressure and normal water level at L4. The reactor has been operating at 102% of rated power for at least 100 days.
- The nominal ANSI/ANS 5.1-1994 decay heat model is assumed with an initial core power of 102%.
- SBO starts with loss of all alternating current (AC) power, which occurs at time zero. Auto bus transfer is assumed to fail.

- Loss of AC power trips reactor, feedwater, condensate and circulating water pumps, and initiates a turbine load rejection.
- The reactor scrams occurs at 2.0 seconds due to loss of power supply to feedwater pumps. When feedwater flow is lost, there is a scram signal with a delay time of 2.0 seconds.
- BPV open on load rejection signal.
- Closure of all Main Steam Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2 after 30 second time delay. The valves are fully closed at 5.0 seconds after signal.
- CRD pumps are not available due to loss of all AC power. The systems available for initial vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:
  - Three Isolation Condensers (ICs)
  - The rest of the safety systems are not credited or they do not actuate during the calculated sequence of events.
- The passive IC system is automatically initiated upon the loss of feedwater pump power buses at 3 seconds, to remove decay heat following scram and isolation and IC drain flow provides initial reactor coolant inventory makeup to the reactor pressure vessel.
- The vessel depressurizes, the vessel and other components inventory remains constant; however, the measured level changes because reactor pressure and liquid temperature changes.
- Other assumptions in Tables 15.2-1, 15.2-2 and 15.2-3, are applied to the TRACG calculation

15.5.5.3 Analysis Results

The results of the system response analyses for the initial core loading documented in Reference 15.5-3 are provided in Reference 15.5-4. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.5-5. A summary is provided in Appendix 15D. As shown in Figures 15.5-10a through 15.5-10e and Table 15.5-10a, during the first 20,000 seconds of depressurization, level is maintained above Level 1. Vessel inventory analysis demonstrates that level remains above Level 1 during the first 72 hours of the transient. Therefore, the requirement for reactor vessel coolant integrity is satisfied. As shown in Table 15.5-10b, considering a constant mass balance, and increased liquid density, the wide range measured level is above 13.5 m (44.3 ft) above vessel zero, which provides adequate margin to L1 ADS analytical limit [11.5 m (37.7 ft) above vessel zero]. The collapsed water level remains well above TAF.

Subsequent to a SBO event, hot shutdown condition can be achieved and maintained by operation of IC systems. Therefore, the requirement for achieving and maintaining hot shutdown condition is met.

15.5-3 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.5-4 GE-_Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337, Class I, Revision 0, ~~Scheduled September~~October 2007.

15.5-5 GE-_Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338, Class I, Revision 0, ~~Scheduled September~~October 2007.

15.5-6 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report", NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non-proprietary), Revision 2, April 2007.

15.5-7 GE Hitachi Nuclear Energy, "TRACG Application for ESBWR," NEDE-33083P Supplement 3, Class III, (Proprietary), December 2007.

15D. EFFECT OF FEEDWATER TEMPERATURE VARIATION

15D.1. INTRODUCTION

Reference 15D-1 presents the transient and accident analyses performed for development of the ESBWR Power – Feedwater Temperature Operating Domain or Map shown in Figure 4.4-1. Section 1.2 of Reference 15D-1 discusses the basis of selecting the events analyzed for 'low' (100% power, FW temperature of 160°C/320°F) and 'high' (85% power, FW temperature of 252.2°C/486°F) feedwater temperature operating states. The events analyzed for these two operating states for the initial core as presented in Section 1.2 of Reference 15D-1 are the following:

(1) Stability

(2) Analyzed AOOs

- a. Loss of Feedwater Heating
- b. Fast Closure of One Turbine Control Valve
- c. Generator Load Rejection With Turbine Bypass
- d. Generator Load Rejection with a Single Failure in the Turbine Bypass System
- e. Inadvertent Isolation Condenser Initiation

(3) Analyzed Infrequent Events

- a. Loss of Feedwater Heating with Failure of Selected Control Rod Run-In
- b. Generator Load Rejection with Total Turbine Bypass Failure
- c. Stuck Open Safety-Relief Valve

(4) Special Events

- a. Main Steam Isolation Valve (MSIV) Closure with Flux Scram
- b. Limiting Anticipated Transient Without Scram (ATWS)
  - i. MSIV Closure – SLC System - Bounding
  - ii. Loss of Condenser Vacuum – SLC System - Bounding
- c. Station Blackout (SBO)

(5) Analyzed LOCA Events for ECCS Performance

- a. GDCS Injection Line Break - Nominal Conditions
- b. GDCS Injection Line Break - Bounding Conditions

(6) Analyzed LOCA Events for Containment Pressure and Temperature

- a. Feedwater Line Break - Nominal Conditions
- b. Feedwater Line Break - Bounding Conditions
- c. Main Steam Line Break - Nominal Conditions

d. Main Steam Line Break - Bounding Conditions

(7) Annulus and Containment Sub-compartment Pressurization

(8) Containment Load Analysis.

Summary results of Stability, LOCA events for Containment Pressure and Temperature, and LOCA events for ECCS Performance are presented in Appendix 4D, and Sections 6.2 and 6.3, respectively. Summaries of other events are presented below.

15D.2. AOO ANALYSES

Table 4.1-3 of Reference 15D-1 presents the comparison of important parameters such as maximum neutron flux, maximum vessel pressure, maximum fractional change in critical power ratio (ΔCPR/ICPR) of the potentially limiting AOO events at 'nominal' (100% power, FW temperature of 215.6°C/420°F), 'high' and 'low' feedwater temperature conditions. The maximum ΔCPR/ICPR values occur during the Loss of Feedwater Heating with no credit for SCRRI/SRI and Inadvertent Isolation Condenser Initiation events. Variation with respect to the feedwater temperature is relatively small. OLMCPR multiplication factors that support operation from 'low' (160°C/320°F) through 'high' (252.2°C/486°F) feedwater temperature operating states are provided.

The results support operation from 'low' through 'high' feedwater temperature operating states.

15D.3. INFREQUENT EVENTS ANALYSES

Table 4.1-4 of Reference 15D-1 presents the comparison of important parameters such as maximum neutron flux, maximum vessel pressure, maximum fractional change in critical power ratio (ΔCPR/ICPR) of the potentially limiting infrequent events at 'nominal', 'high' and 'low' feedwater temperature conditions. Here also, Loss of Feedwater Heating with SCRRI/SRI failure and no credit for high simulated thermal power scram produces the maximum ΔCPR/ICPR value. The consequences are bounded by 1000 rods in transition boiling. The results support operation from 'low' through 'high' feedwater temperature operating states.

15D.4. SPECIAL EVENTS ANALYSES

Table 4.1-5 of Reference 15D-1 presents the maximum vessel bottom pressure during the MSIV Closure with Flux Scram event for 'nominal', 'high' and 'low' feedwater temperature conditions. The maximum pressure remains below the acceptance criterion of 9.481 MPa gauge (1375 psig) for all cases.

Tables 4.1-6 and 4.1-7 of Reference 15D-1 present the important parameters, namely, the maximum vessel bottom pressure, maximum bulk suppression pool temperature, associated containment pressure and peak cladding temperature, of the limiting ATWS events (MSIV Closure and Loss of Condenser Vacuum) for 'nominal', 'high' and 'low' feedwater temperature conditions. The maximum values of all important parameters for all cases remain within the ATWS acceptance criteria presented in the DCD Table 15.5-1.

Table 4.1-8 of Reference 15D-1 presents the theoretical vessel conditions at 72 hours after a Station Blackout (SBO) special event. The collapsed water level in all cases remains above the top of active fuel (TAF) with sufficient margin.

The results of all special events support operation from 'low' through 'high' feedwater temperature operating states.

15D.5. OTHER ANALYSES

Results of Annulus and Containment Sub-compartment Pressurization analyses for 'high' and 'low' feedwater temperature operating states are discussed in Sections 2.8 and 3.8, respectively, of Reference 15D-1. Corresponding results for limiting containment load analyses are presented in Sections 2.9 and 3.9 of Reference 15D-1.

The results support operation from 176.7°C/350°F through 'high' (252.2°C /486°F) feedwater temperature operating states.

15D.6. ANALYSES FOR RELOADS

Based on the results presented in Reference 15D-1, the following AOOs and infrequent events are recommended for reanalysis for each reload since these have the potential of slightly changing the off-rated OLMCPR multiplication factor:

- Loss of Feedwater Heating;
- Closure of One Turbine Control Valve, Fast;
- Generator Load Rejection with Turbine Bypass;
- Inadvertent Isolation Condenser Initiation;
- Loss of Feedwater Heating with SCRRI Failure; and
- Load Rejection with Total Bypass Failure.

However, from the comparison between the Initial and Equilibrium core results reported in Section 3.0 of Reference 15D-2 and the comparison of 'nominal', 'high' and 'low' feedwater temperature results reported in Section 4.1 of Reference 15D-1, the cycle-specific modification to the ESBWR Power – FW temperature operating domain is expected to be minimal.

15D.7. REFERENCES

15D-1 GE Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," NEDO-33338, Class I, October 2007.

15D-2 GE Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analysis," NEDO-33337, Class I, October 2007.