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Subject: **Transmittal of ESBWR Tier 2 DCD Markups Related to Engineered Safety Systems**

The purpose of this letter is to submit markups to the ESBWR DCD, Tier 2 Chapter 6. The changes were previously committed by GEH but were not incorporated into DCD Revision 6. The markup pages are contained in Enclosure 1. The changes are as outlined below:

RAI	Response MFN	Response Date	Status
6.2-13 S01	06-264 S02	August 14, 2007	Paragraph added to Section 6.2.1.2.3 with results updated per RAI 6.2-23S02. (See Enclosure 1)
6.2-64	06-215	July 12, 2006	New 4th paragraph in Section 6.2.1.1.3. (See Enclosure 1)
6.2-65	06-215	July 12, 2006	New 5th paragraph in Section 6.2.1.1.3. (See Enclosure 1)
6.2-69	06-215	July 12, 2006	Changes were made in DCD Rev 6 to Tier 2 Table 6.2-6 rows 16 and 17 and Tier 1 ITAAC Table 2.15.1-2 Item 13 to definitively address the NRC concerns. Maximum and minimum analytical values analyzed for drywell and wetwell volumes, respectively, are specified and the ITAAC ensures the as-built volumes match or are conservative with respect to the containment performance analysis.

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

Sincerely,

Richard E. Kingston

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Vice President, ESBWR Licensing

Enclosure:

1. MFN 10-003 Transmittal of ESBWR Tier 2 DCD Markups Related to Engineered Safety Systems – DCD Tier 2 Sections 6.2.1.1.3 and 6.2.1.2.3.

cc:	AE Cubbage	USNRC (with enclosures)
	JG Head	GEH/Wilmington (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	SC Moen	GEH/Wilmington (with enclosures)
	eDRFSection	0000-0109-8658

Enclosure 1

MFN 10-003

Transmittal of ESBWR Tier 2 DCD Markups

Related to Engineered Safety Systems

DCD Tier 2 Sections 6.2.1.1.3 and 6.2.1.2.3

modules. Once in the suppression pool, the water can be used to maximum advantage for accident mitigation (that is, by restoration of RPV inventory). Figure 5.2-3 shows the location of the spillover holes.

In the event of a pipe break within the DW, the increased pressure inside the DW forces a mixture of noncondensable gases, steam and water through either the PCCS or the vertical/horizontal vent pipes and into the suppression pool where the steam is rapidly condensed. The noncondensable gases transported with the steam and water are contained in the free gas space volume of the WW.

Performance of the pressure suppression concept in condensing steam under water (during LOCA blowdown and SRV discharge) has been demonstrated by a large number of tests, as described in Reference 3B-1.

The SRVs discharge steam through their discharge piping (equipped with quencher discharge device) into the suppression pool. Operation of the SRVs is intermittent, and closure of the valves with subsequent condensation of steam in the discharge piping can produce a partial 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, two nonsafety-related systems are available to provide long-term containment cooling. These systems are FAPCS and Reactor Water Cleanup (RWCU). Heat is removed via either the FAPCS or RWCU nonregenerative heat exchanger(s) to the Reactor Component Cooling Water System (RCCWS) and finally to the Plant Service Water System (PSWS). This FAPCS function is described in Subsection 9.1.3, while for RWCU, this function is described in Subsection 5.4.8. The nonregenerative heat exchanger capacity used for the containment pressure response analysis, is specified in Table 5.4-3.

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 160ºC (320ºF) to 252.2ºC (486ºF) for the bounding Feedwater Line Break (FWLB) and Main Steam Line Break (MSLB) analyses with failure of one SRV. 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). Nuclear Regulatory Commission (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, 6.2-4, and 6.2-11. TRACG is applicable to LOCAs covering the complete spectrum of pipe break sizes, from a small break accident to a DBA, and covering the entire LOCA transient including the blowdown period, the GDCS period and the long-term cooling PCCS period.

The expected operating range of drywell temperature is from 46.1°C (115°F) to 57.2°C (135°F). Cooler initial temperature represents more initial inventory for the noncondensable gases, and consequently higher long-term containment pressure. Therefore, the analyses were performed at 46.1°C (115°F) as given in Table 6.2-6 to ensure conservative peak drywell pressure.

The lower bound on the relative humidity in the drywell is 20% as given in Table 6.2-6. The lower bound value was selected for the analyses, because a lower initial drywell relative humidity results in more noncondensable gases available to be transferred to the wetwell and higher containment pressures following the LOCA.

Containment Design Parameters

Tables 6.2-1 through 6.2-4 provide a listing of key design and operating parameters of the containment system, including the design characteristics of the DW, WW and the pressure suppression vent system and key assumptions used for the design basis accident analysis.

Tables 6.3-1 through 6.3-2 provide the performance parameters of the related ESF systems, which supplement the design conditions of Table 6.2-1, for containment performance evaluation. Table 15.2-23 provides the response time limits for initiation signals used/assumed in accident analyses.

ESBWR Core Decay Heat

The ESBWR core decay heat is generated based on the American National Standards Institute (ANSI)/American Nuclear Society (ANS)-5.1-1994 standard with additional terms to account for a more complete shutdown power assessment. The heat sources considered include decay heat from fission products, actinides and activation products; as well as fission power from delayed and prompt neutrons immediately after shutdown. The effect of neutron capture in fission products is considered. However, initial energy stored in the fuel assembly and heat from metal-water reaction during severe accident are not considered in the decay heat calculations.

The input parameters for the ESBWR decay heat calculation are derived based on the equilibrium core design. Additional safety margins are added to the core parameters in order to bound future cycle variations as well as other fuel product lines with similar parameters. The fuel type assumed in the ESBWR decay heat calculation is GE14E. A constant power irradiation for 3.8 years to reach end-of-cycle exposure of 32 GWd/ST (35.3 GWd/MT) is assumed prior to shutdown. The shutdown mode assumed in the ESBWR decay heat is consistent with a design-basis large break LOCA.

The decay heat modeling for the break LOCA events discussed in this chapter, including the listed fission power after shutdown or decay heat values is consistent, adequate and applicable to the entire LOCA break spectrum due to conservatisms included in its application (Figure 6.2-8c). These values represent the core average decay heat at the end-of-cycle.

Accident Response Analysis

The containment functional evaluation is based upon the consideration of a representative spectrum of postulated accidents, which would result in the release of reactor coolant to the containment. These accidents include:

- Liquid Breaks
  - An instantaneous guillotine rupture of a feedwater line;
  - An instantaneous guillotine rupture of a GDCS line; and
  - An instantaneous guillotine rupture of a vessel bottom drain line.
- Steam Breaks
  - An instantaneous guillotine rupture of a main steamline.

Containment design basis calculations are performed for a spectrum of possible pipe break sizes and the results show that the Double-Ended Guillotine (DEG) pipe break is limiting. Table 6.2-5 summarizes the results of these DEG pipe break calculations. Subsections 6.2.1.1.3.1 through 6.2.1.1.3.5 discuss the results of these calculations. Additional TRACG outputs for the limiting break base case (main steam line break), e.g., the transient air mass profiles in different regions such as the gravity driven cooling system, DW head, and WW airspaces were generated. Also, additional parametric cases were performed to evaluate the impact of various model/plant parameters on the long-term containment pressure. The results of these additional outputs and parametric analyses are detailed in Appendix 6H.

6.2.1.1.3.1 Feedwater Line Break – Nominal Analysis

This analysis initializes the RPV and containment at the base conditions shown in the Nominal Value column of Table 6.2-6. Figure 6.2-6 and 6.2-7 show the TRACG nodalization of the RPV and the containment. Its fundamental structure is an axisymmetric "VSSL" component with 42 axial levels and eight radial rings. The inner 4 rings in the first 21 axial levels represent the RPV; the outer 4 rings in these levels are not utilized in the calculations. Axial levels 22 to 35 represent the DW, suppression pool, WW, and GDCS pools (Figure 6.2-7). Axial levels 36 to 42 subcompartment pressure responses were analyzed with TRACG. The integrity of RSW is discussed in DCD Subsection 3G.1.5.4.2.3.

The break locations have been selected to maximize the mass and energy release into the subcompartment. Since instantaneous double-end guillotine breaks were postulated for all pipe breaks, Leak-Before-Break was not used to limit the break area. The mass and energy release rates are held constant for the analyses. For a feedwater line break, the critical flux is 9.389 x 10^4 kg/(s · m^2), (19230 lb/(s · ft^2)) from either end of the guillotine break, and the total mass release rate from both the RPV end and the reactor shield wall is 2854 kg/s, (6292 lb/s). For a RWCU line break, the critical flux is 4.868 x 10^4 kg/(s · m^2), (9970 lb/(s · ft^2)) from either end of the guillotine break, and the total mass release rate from both the RPV end and the reactor shield wall is 2395 kg/s, (5280 lb/s). Analyzed with TRACG, the peak subcompartment pressure responses were found to be below the design pressure for all postulated pipe break accidents.

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, 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 (Table 6.2-12d, 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. To maximize the energy release to the containment during the blowdown and reflood phases of a LOCA, the following conservative assumptions are used in the analyses.
    - All non-wall heat structures inside the DW and WW are conservatively ignored in the analyses.
    - The DW basemat and the top DW top slab (horizontal heat slabs) are expected to see some steam condensation during the early part of the LOCA. These horizontal heat slabs are conservatively ignored in the analyses.