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MFN 10-035

Docket No. 52-010

January 27, 2010

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

Subject: **Transmittal of ESBWR Tier 2 DCD Chapter 4 Markups Related to Reactor**

The purpose of this letter is to submit ESBWR DCD, Tier 2 Chapter 4 markups that are being incorporated into Revision 7. The changes reflected by the markups are corrections identified by GEH, which consist of the following: 1) corrected a reference in Section 4.4.3.2; 2) deleted reference 4.4-1 in Section 4.4.8 since it is no longer used; and 3) corrected description of reactor coolant heat source during plant startup in Section 4D. The markup pages are contained in Enclosure 1.

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

Sincerely,

Richard E. Kingston

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Vice President, ESBWR Licensing

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NRO

Enclosure:

1. MFN 10-035 Transmittal of ESBWR Tier 2 DCD Chapter 4 Markups Related to Reactor – DCD Tier 2 Sections 4.4.3.2, 4.4.8, and 4D.2.2.1

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	0000-0111-9428

Enclosure 1

MFN 10-003

Transmittal of ESBWR Tier 2 DCD Markups

Related to Engineered Safety Systems

DCD Tier 2 Sections 4.4.3.2, 4.4.8, and 4D.2.2.1

4.4.3.1 Critical Power Evaluations

4.4.3.1.1 Bundle Critical Power Performance Evaluation

The bundle critical power performance results are described in Reference 4.4-12. This reference utilizes full-scale test data to support the development of the critical power correlation for ESBWR. Compliance to steady-state MCPR operating limits is demonstrated for a typical simulation of an equilibrium cycle in Appendix 4A.

4.4.3.1.2 Fuel Cladding Integrity Safety Limit Evaluation

The Fuel Cladding Integrity Safety Limit (FCISL) is defined as 99.9% of the total fueled rods are expected to avoid boiling transition during normal operation and AOOs. Section 6 of Reference 4.4-12 provides a summary of the basis for the representative operating limit MCPR used for the ESBWR to protect the FCISL. Section 5 of Reference 4.4-12 provides the basis for the uncertainties specific to the ESBWR used in this evaluation.

4.4.3.1.3 MCPR Operating Limit Evaluation

The MCPR Operating Limit ΔCPR/ICPR results are described in Section 15.2. The MCPR Operating Limit development including incorporation of the Fuel Cladding Integrity Safety Limit uncertainties is described in Reference 4.4-12.

4.4.3.1.4 MCPR Safety Limit Evaluation

The ESBWR representative MCPR safety limit is in Section 6 of Reference 4.4-12.

4.4.3.2 Void Fraction Distribution Evaluations

The axial distribution of void fractions for an average power channel and a conservative hot channel as predicted by TRACG are given in Table 4.4-2a and Table 4.4-2b. The core average and maximum exit values are also provided. Similar distributions for steam quality are given in Table 4.4-3a and Table 4.4-3b. The axial power distribution used to produce these tables is given in Table 4.4-4a and Table 4.4-4b. The axial void and power distributions for the channel with the highest exit void fraction for the core reference loading pattern (Appendix 4A) are given in Table 4.4-5.

The expected operating void fraction for the ESBWR is within the qualification basis of the void fraction methods. The void fractions in Table 4.4-2a and 4.4-2b are based on TRACG. The hot channel in Table 4.4-2b is a hypothetical channel with a bundle power (radial power) set so as to result in a CPR of 1.20. This hot channel has a maximum void fraction of 0.93. This is conservative compared to the assumed OLMCPR for ESBWR. The void fraction qualification ~~database~~ (References ~~4.4-1 and~~ 4.4-11) contains void fractions in excess of 0.93 and covers the void fraction range expected for normal steady-state operation as well as AOOs. The channel pressure drop qualification is discussed in Subsection 4.4.2.3.5. The core simulator maximum exit void fraction, for the steady-state simulation in Appendix 4A, is 0.89 as shown in Table 4.4-5. The results presented in Tables 4.4-2a to 4.4-5 correspond to the reference equilibrium core discussed in Section 4.3 and Appendix 4A. Similar results corresponding to the initial core, described in Reference 4.4-20, are presented in Reference 4.4-17.

"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 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 ~~GE Nuclear Energy, "Critical Power and Pressure Drop Tests of Simulated 10X10 Bundle Designs Applicable to GE14, NEDC-32874P, Class III (Proprietary), March 2000~~(Deleted).

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, Class I (Non-proprietary), 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, Class I (Non-proprietary), 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, Class I (Non-proprietary), January 1977.

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

This corresponds to Path B in Figure 4D-9.

For the ESBWR at 200 kPa (29 psia), the Zuber Number is of the order of 22, the subcooling number is 22 and the flashing number is 25 and the trajectory corresponding to Path A is followed during the heatup.

4D.2.2 TRACG Analysis of Typical Startup Trajectories

4D.2.2.1 ESBWR Plant Startup

Detailed startup procedures for the ESBWR are developed at a later stage under the guidance of the human factor engineering (Section 18.9 and Reference 4D-19). These procedures are required to observe the rod withdrawal sequence and coolant heatup rate limits, as determined in the analyses reported in the following sections and Reference 4D-20.

The startup process is expected to generally follow the established procedure from the Dodewaard plant. The Dodewaard plant started up for 22 cycles of operation without any problems related to flow or power oscillations.

Figure 4D-10 shows the stages of the startup process. In the De-aeration Period, the reactor coolant is de-aerated by drawing a vacuum on the main condenser and reactor vessel using mechanical vacuum pumps with the steam drain lines open. The reactor coolant is heated up to between 80°C (176°F) and 90°C (194°F) with ~~the RWCU/SDC~~ auxiliary heat~~er~~ and decay heat. The reactor pressure is reduced to about 50 kPa (7.25 psia) to 60 kPa (8.7 psia). Following de-aeration control rods are withdrawn to criticality with the Main Steam Isolation Valves (MSIVs) either left open or closed. The analysis here presents simulation results with the equilibrium core documented in Reference 4D-27 and keeping MSIVs closed. (Plant startup with initial core and MSIVs open is documented in Reference 4D-20.) Startup period is initiated by pulling groups of control rods to criticality. Fission power is used to heat the reactor water, while maintaining the water level close to the top of the separators but well below the steam lines. Steaming at the free surface starts to pressurize the reactor vessel. The core region remains subcooled due to the large static head in the chimney and separators.

As the reactor heats up and pressurizes, the RWCU/SDC system heat exchangers are used to control the downcomer temperature, enhance coolant flow and reduce lower plenum stratification. The MSIVs are reopened at the end of the Startup Period, when pressure reaches 6.3 MPa (~ 914 psia). Subsequently, the turbine bypass valves are used to control pressure. The reactor power is increased and preparations made to roll the turbine.

4D.2.2.2 TRACG Calculations for Simulated Startup Scenarios

The startup transient for the ESBWR is simulated with TRACG. These TRACG calculations are performed with imposed core power, without activating the kinetics model. This is valid as long as there are no feedbacks from oscillations in the core void fraction during the startup transient. This assumption is validated as part of the calculation. The calculation is initiated at the end of the de-aeration period with the steam dome pressure at 52 kPa (7.54 psia) and RPV water at 82°C (~ 180°F). The water level is maintained near the top of the separators. The MSIVs are closed to isolate the RPV. To simplify comparisons, the power level is maintained constant until the pressure reached 6.3 MPa (914 psia). Subsequently, the MSIVs are opened and the power level is increased in steps to achieve rated pressure at 300 MWt (6.67% of rated power).