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Subject: **Transmittal of ESBWR Tier 2 DCD Markups Supporting Response to RAI 6.2-148 S03 Related to Engineered Safety Systems**

The purpose of this letter is to submit markups to the ESBWR DCD, Tier 2 Chapters 1, 5, and 6. The changes are necessary for the implementation of commitments made in response to the Reference 1 Request for Additional Information (RAI), specifically the reference to NEDE-33564P, which was transmitted to the NRC by Reference 2. The markup pages are contained in Enclosure 1.

An additional markup for DCD Chapter 5 is provided to clarify the operation of the RWCU/SDC pumps, specifically reconciling statements in DCD Section 5.4.8.2.2 with Table 19.2-3.

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

Sincerely,

Richard E. Kingston
Vice President, ESBWR Licensing

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NRO

References:

1. MFN 10-048, Letter from U.S. Nuclear Regulatory Commission to Jerald G. Head, *Request for Additional Information Letter No. 406 Related to ESBWR Design Certification Application,* January 21, 2010
2. MFN 08-344 Supplement 4, Letter from Richard E. Kingston to U.S. Nuclear Regulatory Commission, *Transmittal of NEDE-33564P, "Leakage Detection Instrumentation Confirmatory Test for the ESBWR Wetwell-Drywell Vacuum Breakers," in Support of Response to RAI Number 6.2-148 S03*, March 25, 2010

Enclosure:

1. MFN 08-344 Supplement 5 Transmittal of ESBWR Tier 2 DCD Markups Supporting Response to RAI 6.2-148 S03 Related to Engineered Safety Systems – ESBWR DCD Tier 2 Markups

cc:	AE Cubbage	USNRC (with enclosures)
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	eDRFSection	0000-0117-5236

Enclosure 1

MFN 08-344 Supplement 5

Transmittal of ESBWR Tier 2 DCD Markups

Supporting Response to RAI 6.2-148 S03

Related to Engineered Safety Systems

ESBWR DCD Tier 2 Markups

Table 1.6-1

Referenced GE / GEH Reports

Report No.	Title	Section No.
NEDE-33516P	[*GE Hitachi Nuclear Energy, "ESBWR Qualification Plan Requirements for a 72-Hour Duty Cycle Battery," NEDE-33516P, Class III (Proprietary), Revision 2, December 2009.*]*	3.11
NEDE-33536P NEDO-33536	[*GE-Hitachi Nuclear Energy, "Control Building and Reactor Building Environmental Temperature Analysis for ESBWR," NEDE-33536P, Class III (Proprietary), Revision 0, December 2009, NEDO-33536, Class I (Non-proprietary), Revision 0, December 2009.*]*	3H
NEDE-33572P NEDO-33572	GE Hitachi Nuclear Energy, "ESBWR PCCS Condenser Structural Evaluation," NEDE-33572P, Class II (Proprietary), Revision 0, March 2010; NEDO-33572, Revision 0, Class I (Non-proprietary), March 2010.	3G.1, 3.8, 5.4, 6.2
NEDE-33564P NEDO-33564	GE Hitachi Nuclear Energy, "Leakage Detection Instrumentation Confirmatory Test for the ESBWR Wetwell-Drywell Vacuum Breakers," NEDE-33564P, Class II (Proprietary), Revision 0, March 2010; NEDO-33564, Revision 0, Class I (Non-proprietary), March 2010.	6.2

* References that are bracketed and italicized with an asterisk following the brackets are designated as Tier 2*. Prior NRC approval is required to change Tier 2* information.

5.4.8.2.2 System Description

In conjunction with the heat removal capacity of either the main condenser and/or the isolation condensers, the RWCU/SDC system can reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below boiling at atmospheric pressure in less than one day (see Table 5.4-3). The system is also designed to control the reactor temperature reduction rate.

The system can be connected to nonsafety-related standby AC power (diesel-generators), allowing it to fulfill its reactor cooling functions during conditions when the preferred power is not available.

The shutdown cooling function of the RWCU/SDC system provides decay heat removal capability at normal reactor operating pressure as well as at lower reactor pressures.

The redundant trains of RWCU/SDC permit shutdown cooling even if one train is out of service; however, cooldown time is extended when using only one train.

In the event of loss of preferred power, the RWCU/SDC system, in conjunction with the isolation condensers, is capable of bringing the RPV to the cold shutdown condition in a day and a half, assuming the most limiting single active failure, and with the isolation condensers remove the initial heat load. Refer to Subsection 5.4.8.1.2 for a description of the RWCU/SDC pump motor ASD and its operation for shutdown cooling.

In the event of a severe accident resulting in fuel failure, train A of the RWCU/SDC system can be cross-connected to the FAPCS suppression pool suction and the FAPCS containment cooling line to provide containment cooling capabilities. This will allow containment cooling while maintaining the contaminated water inside the reactor building. In this condition the RWCU/SDC system has the capability to return cooled suppression pool water to the reactor vessel through the RWCU mid-vessel suction to preclude using the feedwater injection flowpath, which exits the reactor building.

System Operation

The licensee's operational program and operating procedures for RWCU/SDC (see Sections 13.4 and 13.5, respectively) will incorporate the operational program requirements from the PRA Risk Insights and Assumptions, Table 19.2-3. The modes of operation used for the design of the shutdown cooling function are described below:

Normal Plant Shutdown — The operation of the RWCU/SDC system at high reactor pressure reduces the plant reliance on the main condenser or ICS. The entire cooldown is controlled automatically. During the initial phase of reactor shutdown, the RWCU/SDC pumps operate at reduced speed with the pumps and system configuration aligned to provide a moderate system flow rate and control the cooldown rate to less than the maximum RPV cooling rate allowed. ~~One or both trains of RWCU/SDC may be operated during the early phase of reactor shutdown and cooldown.~~ As cooldown proceeds and RWCU/SDC removes a larger portion of the reactor decay heat, total RWCU/SDC system flow is increased.

In each RWCU/SDC train, the bypass line around the RHX, and the bypass line around the demineralizer are opened to permit increased pump speed and obtain the quantity of system flow required to achieve the process state needed during the shutdown cooling mode. Flow continues

- The containment structure shall withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.
- The containment structure shall accommodate flooding to a sufficient depth above the active fuel to maintain core cooling and to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.
- The containment structure shall be protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes, which could endanger the integrity of the containment.
- The containment structure shall direct the high energy blowdown fluids from postulated LOCA pipe ruptures in the DW to the pressure suppression pool and through the PCCS condensers.
- The containment system shall allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the containment structure to confirm the leak-tight integrity of the containment.
- The Containment Inerting System establishes and maintains the containment atmosphere to $\leq 3\%$ by volume oxygen during normal operating conditions to ensure inert atmosphere operation.
- PCCS shall remove post-LOCA decay heat from the containment for a minimum of 72 hours, without operator action, to maintain containment pressure and temperature within design limits.

6.2.1.1.2 Design Features

The containment structure is a reinforced concrete cylindrical structure, which encloses the Reactor Pressure Vessel (RPV) and its related systems and components. Key containment components and design features are exhibited in Figures 6.2-1 through 6.2-5. The containment structure has an internal steel liner providing the leak-tight containment boundary. The containment is divided into a DW region and a WW region with interconnecting vent system. The functions of these regions are as follows:

- The DW region is a leak-tight gas space, surrounding the RPV and reactor coolant pressure boundary, which provides containment of radioactive fission products, steam, and water released by a LOCA, prior to directing them to the suppression pool via the DW/WW Vent System. A relatively small quantity of DW steam is also directed to the PCCS during the LOCA blowdown.
- The WW region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water to absorb energy by condensing steam from SRV discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The flow path to the WW is designed to entrain radioactive materials by routing fluids through the suppression pool during and following a LOCA. The gas space above the suppression pool is leak-tight and sized to collect and retain the DW gases following a pipe break in the DW, without exceeding the containment design pressure.

The DW/WW Vent System directs LOCA blowdown flow from the DW into the suppression pool.

The containment structure consists of the following major structural components: RPV support structure (pedestal), diaphragm floor separating DW and WW, suppression pool floor slab, containment cylindrical outer wall, cylindrical vent wall, containment top slab, and DW head. The containment cylindrical outer wall extends below the suppression pool floor slab to the common basemat. This extension is not part of containment boundary; however, it supports the upper containment cylinder. The reinforced concrete basemat foundation supports the entire containment system and extends to support the RB surrounding the containment. The refueling bellows, which is treated as a mechanical component, is an all steel, permanent installation with primary and secondary seals that are fabricated from stainless steel for corrosion resistance. The refueling bellows extends from the lower flange of the reactor vessel to the interior of the reactor cavity, and provides a 360° structural barrier to prevent leakage from the reactor cavity into the DW. This extension is also not part of the containment boundary, however, it provides a Seismic Category I seal between the upper DW and reactor well during a refueling outage.

The design parameters of the containment and the major components of the containment system are given in Tables 6.2-1 through 6.2-4. A detailed discussion of their structural design bases is given in Section 3.8.

Refueling Cavity Bellows Seal

As shown in Figure 6.2-35, the RPV is fabricated to include a refueling bellows skirt, and the refueling bellows is installed as a module that is welded to the lower horizontal flange of the RPV refueling bellows skirt. The connections to the drywell bulkhead are also fully welded using AWS standards. The final assembly contains welded connections that provide a permanent barrier to leakage and can only leak if there is a through-wall defect. A spring-loaded secondary seal is provided to prevent leakage into the drywell in the event a leak occurs through the primary seal. The bellows structure is vertically oriented with a corrugated configuration that is designed to be flexible under differential thermal expansion and seismic motion. The bellows material meets ASME/ASTM standards, and bellows examinations are in accordance with Section III of the ASME BPV Code. The bellows assembly is located below the RPV flange such that it cannot interfere with the removal of core components for refueling. Normally open leak detection connections are located on the dry side of the bellows for continuous leakage monitoring.

Cover plates are provided for the bellows to protect against objects (e.g., a fuel assembly) that may be dropped during refueling. The cover plates remain in place during operation, but are designed to be readily removable. In addition, the bellows assembly has a plate over the vertically oriented bellows, which provides further protection in the event the cover plates are removed for cleaning or inspection. Routine maintenance and inspection of the bellows is performed based on supplier recommendations.

As the bellows configuration is not susceptible to dropped objects, any seal failures would likely be related to normal wear or fatigue. All bellows materials are designed to have a 60-year design life. For shop welds, all welding procedures and welder qualifications are in accordance with Section IX of the ASME Code, with any exceptions approved prior to execution. Any leakage due to fatigue or weld failure would not result in a rapid drain down event and would be easily detectible and isolable. As mentioned above, leak detection is provided on the dry side of the bellows, and a second seal is in place that prevents leakage into the drywell in the event of bellows leakage.

Pool level on the refuel floor is constantly monitored, and alarms are provided in the event level drops. The buffer pool is equipped with safety-related water level sensors that alarm the operators if the pool level is below normal. Operators also have the ability to provide makeup water and suspend refueling operations, as needed. In addition to pool level alarms, the drywell sump provides alarms in the event excess water is present.

Drywell

The DW (Figure 6.2-1) comprises two volumes: (1) an upper DW volume surrounding the upper portion of the RPV and housing the main steam and feedwater piping, GDCS pools (see Figure 6.2-3 for pool arrangement) and piping, PCCS piping, ICS piping, SRVs and piping, Depressurization Valves (DPVs) and piping, DW coolers and piping, and other miscellaneous systems; and (2) a lower DW volume below the RPV support structure housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems and equipment below the RPV, and vessel bottom drain piping.

The upper DW is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV support structure separates the lower DW from the upper DW. There is an open communication path between the two DW volumes via upper DW to lower DW connecting vents, built into the RPV support structure. Penetrations through the liner for the DW head, equipment hatches, personnel locks, piping, electrical and instrumentation lines are provided with seals and leak-tight connections.

The DW is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the DW, and also the negative differential pressures associated with containment depressurization events, when the steam in the DW is condensed by the PCCS, the GDCS, the FAPCS, and cold water cascading from the break following post-LOCA flooding of the RPV.

For a postulated DBA, the calculated DW pressure in Table 6.2-5 is below the design value shown in Table 6.2-1. The structure stresses are evaluated in Section 3G.5 considering the DW fluid temperature transients for multiple break locations.

Three vacuum breakers are provided between the DW and WW. The vacuum breaker is a process-actuated valve, similar to a check valve (Figure 6.2-28). The purpose of the DW-to-WW vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the DW and the WW, and the DW structure and liner, and to prevent back-flooding of the suppression pool water into the DW. The vacuum breaker is provided with redundant proximity sensors to detect its closed position. One out of the three vacuum breakers is required to perform vacuum relief function. The third vacuum breaker provides redundancy while the second vacuum breaker provides single failure protection for opening. On the upstream side of each vacuum breaker, pneumatically operated fail-as-is safety-related isolation valves are provided to isolate a leaking (not fully closed) or stuck open vacuum breaker. During a LOCA, when the vacuum breaker opens and allows the flow of gas from WW to DW to equalize the DW and WW pressure and subsequently does not fully close as detected by the proximity sensors, a control signal closes the upstream isolation valve to prevent bypass leakage through the vacuum breaker and therefore maintain the pressure suppression capability of the containment.

In addition to the proximity sensors, there are temperature sensors located on and in the vacuum breaker/vacuum breaker isolation valve assembly. See Figure 6.2-28 for approximate temperature sensor locations and sensor terminology. These sensors will detect a rise in temperature between the vacuum breaker and the end of the penetration on the wetwell side due to the hot DW gas leaking past a not fully closed vacuum breaker. When the difference between the cavity temperature, T_{cavity}, and the wetwell temperature, T_{ww}, exceeds a fraction of the difference between the drywell temperature, $T_{dw(1)}$, and wetwell temperature, T_{ww}, a signal is sent to the vacuum breaker isolation valve to close. The bulk drywell temperature, $T_{dw(2)}$, (Figure 6.2-28) is measured separately from the vacuum breaker/vacuum breaker isolation valve assembly and is used to detect LOCA conditions and acts as a permissive to allow the vacuum breaker isolation valve logic to function.

The corresponding bypass leakage area that the temperature sensors will detect to close a vacuum breaker isolation valve is a maximum analytical limit of 0.6 cm^2 $(A/\sqrt{K})$ (Reference 6.2-15). Closing each vacuum breaker isolation valve at this bypass leakage assures the analytical limit of 2 cm^2 $(A/\sqrt{K})$ of total bypass leakage will not be exceeded in the unlikely scenario of three vacuum breakers not fully closing. This scenario assumes more than one single failure will occur which is beyond design basis accident requirements.

Each vacuum breaker isolation valve logic subsystem is located in physically separate divisional rooms or compartments that have appropriate fire barriers between them. The isolation valve can also be manually opened or closed. For more discussion on the logic control of the vacuum breaker isolation valves, see Subsection 7.3.6. The design WW-to-DW pressure difference and the vacuum breaker opening differential pressure are given in Table 6.2-1.

The vacuum breaker and vacuum breaker isolation valves are protected from pool swell loads by structural shielding/debris screen designed for pool swell loads determined based on the Mark II/III containment design. Both valves are located in the DW and connected to the WW gas space by a penetration through the diaphragm floor. The structural shielding/debris screen is located in the WW gas space at the inlet side of the penetration.

A safety-related PCCS is incorporated into the design of the containment to remove decay heat from DW following a LOCA. The PCCS uses six elevated heat exchangers (condensers) that are an integral part of the containment boundary located in large pools of water outside the containment at atmospheric pressure to condense steam that has been released to the DW following a LOCA. This steam is channeled to each of the condenser tube-side heat transfer surfaces where it condenses and the condensate returns by gravity flow to the GDCS pools. Noncondensable gases are purged to the suppression pool via vent lines. The PCCS condensers are an integral part of the containment boundary, do not have isolation valves, and start operating immediately following a LOCA. These low pressure PCCS condensers provide a thermally efficient heat removal mechanism. No forced circulation equipment is required for operation of the PCCS. Steam produced, due to boil-off in the pools surrounding the PCCS condensers, is vented to the atmosphere. There is sufficient inventory in these pools to handle at least 72 hours of decay heat removal. The PCCS is described and discussed in detail in Subsection 6.2.2.

6.2.6.5 (Deleted)

6.2.7 Fracture Prevention of Containment Pressure Boundary

The reactor containment system includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant postulated accidents.

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 RG 1.26, which assigns correspondence of Group B Quality Standards to ASME Code Section III Class 2.

6.2.8 COL Information

6.2-1-H (Deleted)

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, Revision 1, Class I (Non-proprietary), May 2009.

6.2-8 Moody, F.J. "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels," General Electric Company, Report No. NEDO-21052-A, May 1979.

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.

6.2-10 TRACG Qualification for Simplified Boiling Water Reactor (SBWR), NEDC-32725P, Rev. 1, Vol. 1 and 2, August 2002.

6.2-11 GE Hitachi Nuclear Energy "ESBWR Safety Analysis - Additional Information," NEDE-33440P, Revision 2, Class III (Proprietary), March 2010; NEDO-33440, Revision 1, Class I (Non-proprietary), March 2010.

6.2-12 Idel'chik, I.E., Barouch, A. "Handbook of hydraulic resistance: coefficients of local resistance and of friction," National Technical Information Service, 1960.

6.2-13 SMSAB-02-04, "CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations," Office of Nuclear Regulatory Research, September 2002 (ADAMS Accession Number ML023220288).

6.2-14 GE Hitachi Nuclear Energy, "ESBWR PCCS Condenser Structural Evaluation," NEDE-33572P, Class II (Proprietary), Revision 0, March 2010; NEDO-33572, Revision 0, Class I (Non-proprietary), March 2010.

6.2-15 GE Hitachi Nuclear Energy, "Leakage Detection Instrumentation Confirmatory Test for the ESBWR Wetwell-Drywell Vacuum Breakers," NEDE-33564P, Class II (Proprietary), Revision 0, March 2010; NEDO-33564, Revision 0, Class I (Non-proprietary), March 2010.