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Subject: Response to Portion of NRC Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-145 S01

Enclosure 1 contains the GE Hitachi Nuclear Energy (GEH) response to the subject NRC RAI originally transmitted via the Reference 1 letter and supplemented by an NRC request for clarification in Reference 2. DCD Markups related to this response are provided in Enclosure 2.

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

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

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NRO

References:

1. MFN 07-054, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application*, January 19, 2007
2. E-Mail from Shawn Williams, U.S. Nuclear Regulatory Commission, to George Wadkins, GE Hitachi Nuclear Energy, dated June 13, 2007 (ADAMS Accession Number ML071640395)

Enclosures:

1. MFN 08-359 - Response to Portion of NRC Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-145 S01
2. MFN 08-359 - Response to Portion of NRC Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-145 S01 - DCD Markups

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Enclosure 1

MFN 08-359

Response to Portion of NRC Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application

Containment Systems

RAI Number 6.2-145 S01

NRC RAI 6.2-145 S01:

Appendix A to Section 6.2.1.1.C of the Standard Review Plan (SRP) states, "The Mark II and Mark III acceptance criteria for both the high and low pressure leakage tests shall be a measured bypass leakage which is less than 10% of the capability of the containment ..."

In response to RAI 6.2-145, GE proposed an acceptance criterion for bypass leakage of 2 cm^2 (2.16E-03 ft^2) ($A/\sqrt{K}$) and stated that "DCD Tier 2, Revision 3, Subsection 6.2.1.1.5.1 contains additional information from the latest bounding design basis accident calculations that assume a bypass leakage size of 2 cm^2 (2.16E-03 ft^2) ($A/\sqrt{K}$)."

DCD Tier 2, Rev 3, Section 6.2.1.1.5.1, states that "the bounding design basis accident calculation assumes a bypass leakage of 1 cm^2 (1.08E-03 ft^2), ($A/\sqrt{K}$). Table 6.2-5 shows these results in acceptable containment pressures. Additional bounding design basis accident calculations show also that with a bypass leakage assumption of 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$) the containment pressures continue to be below the design pressure and with a bypass leakage assumption of 14 cm^2 (1.51E-02 ft^2), ($A/\sqrt{K}$) the containment pressures remain below the ultimate pressure capability of the drywell head (1.204 MPag) (see reference 6.2-6) with ample margin." Reliance on the containment ultimate strength for justification of the acceptance criterion is not acceptable to the staff.

A. *It is not clear to the staff whether the design leakage is 1 cm^2 (1.08E-03 ft^2), ($A/\sqrt{K}$) or 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$). Please confirm that the results provided in DCD Tier 2, Rev 3, table 6.2-5 are based on 1 cm^2 (1.08E-03 ft^2), ($A/\sqrt{K}$) bypass leakage area. If so, please provide the containment peak pressure results using 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$) as the assumed bypass leakage, and provide the margin to the containment design pressure.*

B. *GE should provide additional justification of the alternative approach from the SRP guidance related to selection of the acceptance criterion for bypass leakage area. The purpose of the bypass leakage test acceptance criterion is to provide reasonable assurance that the ESBWR bypass leakage area will not exceed the value assumed in the design basis containment peak pressure analysis in between test intervals. GE should propose a bypass leakage test acceptance criteria less that is less than the design basis assumption for bypass leakage and justify that the selected acceptance criterion will provide reasonable assurance that the plant's bypass leakage area will not exceed the value assumed in the plant's safety analyses during postulated design basis accidents.*

GEH Response:

A. The 2.0 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$) is the maximum allowable suppression pool bypass leakage that results in containment peak pressures, for a bounding LOCA, below the containment design pressure. In a presentation to the NRC Staff on March 5, 2008 entitled "Containment Bypass Leakage Capability," GEH discussed the containment response based on the maximum allowable suppression pool bypass leakage and detailed preliminary results. Verified results using this bounding design basis accident allowable bypass area of 2.0 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$), including containment peak pressure and margin, will be provided in DCD Tier 2, Revision 5.

The results shown in DCD Tier 2, Table 6.2-5 are based on a nominal design basis suppression pool bypass leakage of 1.0 cm^2 (1.08E-03 ft^2), ($A/\sqrt{K}$).

B. Suppression pool bypass leakage may be quantified and measured by performing a local leakage rate test on a 24 month frequency and an overall suppression pool bypass leakage test on the same frequency as the Integrated Leakage Rate Test (ILRT). These test frequencies are similar to the following operating BWRs with Mark II containments: Columbia Generating Station, Nine Mile Point Unit 2, Susquehanna Units 1 and 2 and Limerick Units 1 and 2.

Both the local leak rate test and the suppression pool bypass leakage test provide assurance that the suppression pool bypass leakage is maintained within the allowable value between tests. The acceptance criteria for both are in DCD Tier 2, Subsection 6.2.1.1.5.4.3.

The suppression pool bypass leakage test will quantify and measure bypass leakage to ensure the bypass leakage is maintained within limits over the 60-year life of the plant.

The proposed acceptance criteria for the suppression pool bypass leakage test ($A/\sqrt{K}$) is less than or equal to 1.0 cm^2 (1.08E-03 ft^2), which is < 50% of the design basis bypass leakage area.

NUREG-0800 (Standard Review Plan) Section 6.2.1.1.C, "Pressure Suppression Type BWR Containments," Appendix A requires preoperational and periodic suppression pool bypass leakage tests. NUREG-0800 also recommends bypass testing acceptance criteria for existing BWR Mark I, Mark II, and Mark III pressure suppression containments. The capability for steam bypass for small primary system breaks in the Mark I, II, and III containment design are as follows: the Mark I design is of the order of 18.6 cm^2 (0.02 ft^2), the capability of the Mark II containment is approximately 46.5 cm^2 (.05 ft^2), and the Mark III design has a capability of $A/\sqrt{K}$ = 929 cm^2 (1 ft^2). The Standard Review Plan, Appendix A recommends that Mark II and Mark III acceptance criteria for suppression pool bypass leakage tests shall be a measured bypass leakage which is less than 10% of the capability of the containment.

General Electric established the 10% of containment capability during licensing of the initial pressure suppression containments in the early 1970s for BWRs with active ECCS systems. The 10% of containment capability was intended to leave sufficient margin for increases in bypass leakage between outages. This was due in part to the limited amount of actual field-testing experience and data and the large number of penetrations through the diaphragm floor of the Mark II containment. This criterion allows a 90% margin for additional bypass leakage between outages. Existing data indicates that there is little increase in bypass leakage for Mark II containments even over intervals as long as 10 to 15 years and that the actual measured bypass leakage is a small fraction of the design capability.

The acceptance criteria for the suppression pool bypass leakage test for ESBWR is less than or equal to 50% of the design basis suppression pool bypass leakage area of 2.0 cm^2 (2.16E-03 ft^2). The calculated suppression pool bypass area will be reported at the upper 95% confidence level to account for measurement errors. This still allows a 50% margin to the design basis suppression pool bypass leakage area, which ensures an adequate margin to account for bypass leakage increases between tests while not imposing an undue regulatory burden on plant owners.

The acceptance criteria for the individual local leakage rate test of the vacuum breakers and vacuum breaker isolation valves is less than or equal to 15% of the design basis bypass leakage area. The acceptance criteria for the total leakage rates of all the vacuum breaker/vacuum breaker isolation valve pairs on a maximum pathway basis is less than or equal to 35% of the design basis bypass leakage area. These tests will quantify and measure any degradation of suppression pool bypass leakage between refueling outages when the suppression pool bypass test is not conducted.

The ESBWR has been uniquely designed to minimize the potential suppression pool bypass leakage paths. The ESBWR is designed with a reinforced, lined concrete pressure suppression containment structure. This structure encloses the reactor pressure vessel (RPV), including related systems and components and incorporates an internal steel liner that provides a leak-tight containment boundary. Similar to the existing BWR Mark II containments, the ESBWR is divided into a drywell and wetwell region by a diaphragm floor with an interconnecting vent system. However, there are significant differences that make the ESBWR unique.

The diaphragm floor is a composite structure consisting of a plate steel drywell liner with full penetration welds, a concrete slab, and a wetwell stainless steel liner with full penetration welds. Preservice and inservice inspections of the weld integrity of the liner welds and penetration welds through the diaphragm floor will be conducted to ensure leak tightness.

The ESBWR design has minimized the number of penetrations across the diaphragm floor to minimize bypass leakage paths. Existing BWR Mark II containments have over one hundred penetrations through the diaphragm floor with varying outside diameters of 24 inch and 28 inch. The ESBWR containment has only thirteen penetrations through the diaphragm floor with outside diameters of 24 inch (3), 10 inch (6), and 1 inch (4). ESBWR Main Steam Safety Relief Valve (SRV) discharge lines are routed through the containment vent walls and are seal welded at the vent wall penetrations. Bypass leakage across the diaphragm floor penetration welds and plate welds is highly improbable.

The vacuum breakers were developed under SBWR program and a prototype vacuum breaker was built and qualified per the guidance in IEEE 323. Testing was completed in July 1994 and details were issued to the NRC as part of response to RAI 900.62 issued by letter MFN-155-94 dated December 15, 1994 under Docket STN 52-004. ESBWR uses the identical vacuum breakers, and letter MFN 06-127 transmitted SBWR vacuum breaker test program report to NRC staff for review.

The only credible suppression pool bypass leakage path is through the three 24 inch vacuum breakers. The ESBWR uses a uniquely designed leak-tight vacuum breaker, which incorporates both a non-metallic seat and a hard seat. The vacuum breaker isolation valve meets the requirements of Class 6 leakage per ANSI FCI-70-2-2003.

This unique ESBWR diaphragm floor design gives assurance that the suppression pool bypass leakage test acceptance criteria of <50% of the design basis suppression pool bypass leakage area measured at the 95% confidence level can be met over the 60 year life of the plant with sufficient operating margin.

DCD Impact:

DCD Tier 2, Subsections 6.2.1.1.5.1, 6.2.1.1.5.3.2, 6.2.1.1.5.4, 6.2.1.1.5.4.2, 6.2.1.1.5.4.3, and 6.2.1.1.5.4.4, and Reference 6.2-6, will be revised as shown in the attached markup. Additional DCD Tier 2 revisions using the bounding design basis accident allowable bypass area of 2.0 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$), including containment peak pressure and margin, will be provided in Revision 5.

Enclosure 2

MFN 08-359

Response to Portion of NRC Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application

Containment Systems

RAI Number 6.2-145 S01

DCD Markups

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

6.2.1.1.4 Negative Pressure Design Evaluation

During normal plant operation, the inerted WW and the DW volumes remain at a pressure slightly above atmospheric conditions. However, certain events could lead to a depressurization transient that can produce a negative pressure differential in the containment. A DW depressurization results in a negative pressure differential across the DW walls, vent wall, and diaphragm floor. A negative pressure differential across the DW and WW walls means that the RB pressure is greater than the DW and WW pressures, and a negative pressure differential across the diaphragm floor and vent wall means that the WW pressure is greater than the DW pressure. If not mitigated, the negative pressure differential can damage the containment steel liner. The ESBWR design provides the vacuum relief function necessary to limit these negative pressure differentials within design values. The events that may cause containment depressurization are:

- Post-LOCA DW depressurization caused by the ECCS (GDCS, CRD, and so forth) flooding of the RPV and cold water spilling out of the broken pipe or cold water spilling out of broken GDCS line directly into DW.
- The DW sprays are inadvertently actuated during normal operation or during post-LOCA recovery period.
- The combined heat removal of the ICS and PCCS exceeds the rate of decay heat steam production.

Drywell depressurization following a LOCA is expected to produce the most severe negative pressure transient condition in the DW. The results of the MSL break analysis show that the containment does not reach negative pressure relative to the RB, and the maximum wetwell-drywell differential pressure is within the design capability. This calculation assumes one available wetwell-drywell vacuum breaker with an area of 0.2 m^2, (2.16 ft^3), which is conservative with respect to the planned installed vacuum breaker area. An evaluation of the effect of drywell spray on containment integrity for a main steam line break and a feedwater line break was performed to determine the maximum negative differential pressures (drywell to wetwell, and drywell to reactor building). This evaluation assumed that a drywell spray flow rate of 454 m^3/hr (2000 gpm) at a temperature of 293°K is initiated at the worst possible moment for a drywell spray, at the point in time when there is low intert gas content in the DW relative to the WW (i.e., when the drywell pressure has peaked just prior to the opening of the drywell-wetwell vacuum breakers), and verified that the maximum negative differential pressures remain within the design criteria. For additional conservatism and to account for uncertainties in the design of the drywell spray piping system, including drywell spray flow limiting design features, a value of 227 m^3/hr (1000 gpm) has been established as the maximum design operating limit (see Subsection 9.1.3).

6.2.1.1.5 Steam Bypass of Suppression Pool

6.2.1.1.5.1 Bypass Leakage Area in Design Basis Accident

The concept of the pressure suppression reactor containment is that any steam released from a pipe rupture in the primary system is condensed by the suppression pool, and thus, does not produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. If a leakage path were to exist between the drywell and the suppression pool (wetwell) gas space, the leaking steam would produce undesirable pressurization of the containment. The ~~bounding~~ nominal design basis accident calculations assumes a suppression pool bypass leakage of 1 cm^2 (1.08E-03 ft^2), ($A/\sqrt{K}$). Table 6.2-5 shows that this results in acceptable containment pressures. ~~Additional~~ A bounding design basis accident calculation ~~show also that~~ with the maximum allowable~~a~~ suppression pool bypass leakage assumption of 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$) results in ~~the~~ containment pressures ~~continue to be~~ below the design pressure and with a suppression pool bypass leakage assumption of 14 cm^2 (1.51E-02 ft^2), ($A/\sqrt{K}$) the containment pressures remain below the ultimate pressure capability of the drywell head (1.204 MPag) (see Reference 6.2-6) with ample margin.

6.2.1.1.5.2 Suppression Pool Bypass During Severe Accidents

See Chapter 19 for discussion on Suppression Pool Bypass During Severe Accidents.

6.2.1.1.5.3 Justification for Deviation From SRP Acceptance Criteria

6.2.1.1.5.3.1 Actuation of PCCS

The provision of automatic PCCS design meet the intent of the SRP (Appendix A to SRP Subsection 6.2.1.1.C) for automatic actuation of sprays, without the use of a containment spray system. The SRP states that the wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell to quench steam bypassing the suppression pool. However, in determining maximum allowable steam bypass leakage area for ESBWR design, analyses take credit for PCCS operation immediately following LOCA initiation.

The PCCS is considered adequate to provide mitigation for consequences due to steam bypass leakage during a LOCA event.

6.2.1.1.5.3.2 Vacuum Valve Operability Tests

Section B.3.b of Appendix A to SRP Subsection 6.2.1.1.c specifies that vacuum valves should be operability tested at monthly intervals to assure free movement of the valves. Operability tests are conducted at plants of earlier BWR designs using an air actuated cylinder attached to the valve disk. The air actuated cylinders have been found to be one of the root causes of vacuum breakers failing to close. Free movement of the vacuum breakers in the ESBWR design has been enhanced by eliminating this potential actuator failure mode, improving the valve hinge design and selecting materials which are resistant to wear and galling. Therefore, monthly testing is not performed for these vacuum breakers. However, the vacuum breakers are tested for free movement and leakage ~~during each outage~~according to Technical Specification requirements.~~.~~

6.2.1.1.5.4 Bypass Leakage Tests and Surveillance

There are provisions for leakage tests and surveillances to determine suppression pool bypass leakage, and to ensure that leakage does not substantially increase over the plant life.

Pre-operational and periodic suppression pool bypass tests are performed at the same frequency as the Integrated Leakage Rate Test (ILRT). Pre-operational and periodic local leakage rate testing of the vacuum breakers and vacuum breaker isolation valves are also performed. These tests and surveillances will quantify, measure, and/or detect any degradation, and provide assurance that the suppression pool bypass leakage is maintained within the allowable value between tests.~~There is a provision for leakage tests and surveillance to provide assurance that suppression pool bypass leakage is not substantially increased over the plant life. This includes a pre-operational and periodic local leak rate testing of vacuum breakers and, a periodic visual inspection of drywell to wetwell penetrations.~~

6.2.1.1.5.4.1 (Deleted)

6.2.1.1.5.4.2 Local Leak Rate Testing of Drywell to Wetwell

Pre-operational and periodic visual examinations of the drywell to wetwell penetrations are performed in accordance with inservice inspection and/or inservice testing requirements. A suppression pool bypass test is performed to detect leakage from the drywell to the wetwell on the same frequency as the ILRT. Local leakage rate tests of the individual vacuum breakers and vacuum breaker isolation valves are performed according to Technical Specification requirements. The acceptance criteria are specified in Subsection 6.2.1.1.5.4.3.~~A pre-operational and post-operational visual inspection of drywell to wetwell penetrations and local leak rate testing of vacuum breakers is performed to detect leakage from the drywell to wetwell. This test is performed at each refueling outage. A low-pressure test is not conducted since the vacuum breakers are the only credible source for bypass leakage; other existing penetrations are pipe connections that are welded and cannot physically leak. [JGD113][AS114]The acceptance criteria are specified in Subsection 6.2.1.1.5.4.3.~~

6.2.1.1.5.4.3 Acceptance Criteria for Leakage Tests

NUREG-0800, 6.2.1.1.c Draft 1996, Appendix A, Steam Bypass, specifies acceptance criteria for drywell/wetwell steam bypass testing for Mark I, II and III containments. It states that alternative criteria can be proposed for review by the NRC staff. For ESBWR an alternate criteria is proposed, to:

- Provide a drywell/wetwell interface, sufficiently leak tight, to assure the containment performs the intended function of containment of radioactivity.
- Provide flexibility for the licensee in conducting tests.
- Account for degradation in performance between tests.
- Account the uncertainties in test measurement.

The acceptance criteria for the suppression pool bypass test is a calculated bypass leakage area ($A/\sqrt{K}$) that is less than 50% of the bounding design basis accident allowable bypass area, which is 2.0 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$). The calculated bypass leakage area is calculated at the upper 95% confidence level to account for instrument inaccuracies and uncertainties. The acceptance criteria for the individual vacuum breaker and vacuum breaker isolation valve local leakage rate tests is less than or equal to 15% of the bounding design basis accident allowable bypass area, and the acceptance criteria for the total leakage of all three vacuum breaker/vacuum breaker isolation valve pairs on a maximum pathway basis is less than or equal to 35% of the bounding design basis accident allowable bypass area.

Local leakage rate measurement inaccuracies are typically less than or equal to 2% of the full scale flow range. This value is insignificant when compared to the margins between the local leakage rate acceptance criteria and the leakage equivalent to the bounding design basis accident allowable bypass area.~~The criteria specified for Mark II and III containments is a fraction of the analytical leakage capability. The fraction is judged small enough to cover degradation in performance between tests and uncertainties in test measurement. For ESBWR, an alternate acceptance criteria will be applied. The ability of the containment to tolerate degraded (increased) leakage up to ultimate strength has been determined to be more than a factor of 5 above the design capability (see Subsection 6.2.1.1.5.1). This adequately bounds potential degradation between test intervals. The uncertainty in the test measurement will be quantified and applied to the acceptance criteria. The acceptance criteria will be the leakage analytically required to keep the containment below design pressure, 2 cm2 (2.16E-03 ft2), (A/√K). The uncertainties associated with the specific test procedure and equipment applied will be determined by the licensee and added to the measured leakage prior to comparison against the acceptance criteria.[JGD118]~~

6.2.1.1.5.4 Surveillance Test

A visual inspection will be conducted to detect possible leak paths at each refueling outage. Each vacuum ~~relief valve~~ breaker and associated piping will be checked to determine that it is clear of foreign matter. Also, at this time each vacuum breaker will be tested for free disk movement.

6.2.1.1.5.5 Vacuum ~~Relief~~ Breaker Valve and Isolation Valve Instrumentation and Tests

6.2.1.1.5.5.1 Position Indicators, Temperature Sensors, and Alarms

Redundant position indicators are placed on vacuum breakers with redundant indication and an alarm in the control room. The vacuum breaker position indicator system is designed to provide the plant operators with continuous surveillance of the vacuum breaker position. The vacuum ~~relief valve~~breaker position indicator system has adequate sensitivity to detect an ~~total valve~~ open~~ing~~ vacuum breaker~~, for all valves, that is less than the design bypass capability, discussed in Subsection 6.2.1.1.5.4~~.

Redundant temperature sensors are placed within the cavity created by the vacuum breaker and vacuum breaker isolation valve assembly, and in close proximity to the vacuum breaker outlets. The temperature sensor system is designed to provide detection of a leaking vacuum breaker during a LOCA.

6.2.1.1.5.5.2 Vacuum Breaker Valves and Isolation Valves Operability Tests

The vacuum ~~relief valves~~ breakers will be tested for free movement ~~during each refueling outage~~according to Technical Specification requirements. Vacuum breaker isolation valves will be tested as specified in Table 3.9-8.

6.2.1.1.6 Suppression Pool Dynamic Loads

During a postulated LOCA, DW-to-WW flow of gas and steam/water mixture produces hydrodynamic loading conditions on the suppression pool boundary. Also, SRV flow 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)~~ESBWR Certification Probabilistic Risk Assessment, NEDO-33201, Rev. 1, February 2006.

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

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

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