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U.S. Nuclear Regulatory Commission
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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 321 Related to the ESBWR Design Certification – Containment Systems – RAI Number 6.2-145 S03**

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 No. 321 (Reference 1). The TS surveillance frequency of 10 years was based upon extensive GEH data on fleet vacuum breaker testing. During discussions with the Staff, the Staff indicated that every 24 months would be a more appropriate frequency, and that once ESBWR operating data indicated that a longer testing frequency was indicated, NRC would entertain approval of a longer testing frequency. GEH response to RAI Number 6.2-145 S03 is addressed in Enclosure 1. DCD, Technical Specification (TS) and TS Bases markups associated with this response are provided in Enclosure 2.

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

Sincerely,

Richard E. Kingston

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Vice President, ESBWR Licensing

Reference:

1. MFN 09-196, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 321 Related To ESBWR Design Certification Application*, dated March 18, 2009

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 321 Related to ESBWR Design Certification Application – Containment Systems – RAI Number 6.2-145 S03
2. Response to Portion of NRC Request for Additional Information Letter No. 321 Related to ESBWR Design Certification Application – Containment Systems – RAI Number 6.2-145 S03 – DCD and Technical Specifications Markups

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Enclosure 1

MFN 08-612, Supplement 1

Response to Portion of NRC Request for Additional Information Letter No. 321 Related to the ESBWR Design Certification

Containment Systems

RAI Number 6.2-145 S03

NRC RAI 6.2-145 S03:

Justify the proposed bypass surveillance test frequency.

In RAI 6.2-145, Supplement 2, the staff requested GEH to justify its requests for (1) drywell to wetwell bypass leakage surveillance criteria of 50 percent of the design bypass leakage capacity and (2) a change in drywell to wetwell bypass leakage surveillance frequency from outage frequency to that of integrated leak rate testing (ILRT).

GEH responses provided in its letters, dated August 6, 2008 and February 10, 2009, provided sufficient justification for the proposed drywell to wetwell bypass leakage surveillance criteria of 50 percent.

However, GEH responses did not provide sufficient justification for the proposed change to the drywell to wetwell bypass leakage surveillance frequency to that of ILRT. In its response, dated February 10, 2009, GEH showed plant-specific data from operating boiling water reactors with Mark I and Mark II containments.

Those plants used these data to request for changes to the drywell to wetwell bypass leakage surveillance test frequency from outage frequency to that of ILRT. GEH's request is not acceptable to the staff because ESBWR does not have such plant-specific data. ESBWR plants may make such requests later with supporting plant-specific data for staff's review.

GEH should provide additional justification its request to change to the drywell to wetwell bypass leakage surveillance frequency to the ILRT frequency or change the ESBWR DCD to state that plants will perform drywell to wetwell bypass leakage surveillance testing at outage frequency.

GEH Response:

The wetwell bypass leakage surveillance frequency will be changed to 24 months.

DCD Impact:

DCD Tier 2, Subsections 6.2.1.1.5.4 and 6.2.1.1.5.4.2, Surveillance Requirement 3.6.1.1.5, and Surveillance Requirement Bases 3.6.1.1.5 will be revised as noted on the attached markups.

Enclosure 2

Response to Portion of NRC Request for Additional Information Letter No. 321 Related to the ESBWR Design Certification

Containment Systems

RAI Number 6.2-145 S03

DCD and Technical Specifications Markups

suppression pool bypass leakage assumption of 1 cm^2 (1.08E-03 ft^2), (A/√K) are presented in Appendix 6I..

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 WW spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the WW 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 Boiling Water Reactor (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 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.

After the ~~P~~pre-operational ~~and periodic~~ suppression pool bypass tests are performed, ~~at~~ a ~~the same~~ periodic test is performed at a frequency of 24 months~~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 quantify, measure, or detect any degradation, and provide assurance that the suppression pool bypass leakage is maintained within the allowable value between tests.

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 DW to WW penetrations are performed in accordance with inservice inspection or inservice testing requirements. A suppression pool bypass test is performed to detect leakage from the DW to the WW at a~~on the same~~ frequency of 24 months~~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.

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 DW/WW 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 DW/WW 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.

6.2.1.1.5.4.4 Surveillance Test

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

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.5	Verify overall suppression pool bypass leakage is ≤ 50% of design basis A/√K.	~~10 years~~ ~~AND~~ ~~48 months following a test with leakage greater than the leakage limit~~ ~~AND~~ 24 months ~~following two consecutive tests with leakage greater than the leakage limit until two consecutive tests are less than or equal to the leakage limit~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.1.5

Maintaining the pressure suppression function of the containment requires limiting the leakage from the drywell to the wetwell. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the horizontal vent pipes into the wetwell. This SR determines effective overall suppression pool bypass leakage area to ensure that the leakage paths that would bypass the wetwell pressure suppression function are within allowable limits.

Satisfactory performance of this SR can be achieved by establishing a known initial differential pressure (≥ 2.0 psid) between the drywell and the wetwell and verifying that the suppression pool bypass leakage equivalent to an area ≤ 50% of the bounding design basis value A/√K of 2.0 cm^2 (2.16E-03 ft^2).

The overall suppression pool bypass leakage test is performed every ~~10 years~~24 months. The Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage~~ and the desire to perform the test in conjunction with the 10 CFR 50, Appendix J, Type A test. One test failure increases the Surveillance frequency to 48 months. Two consecutive test failures, however, would indicate unexpected degradation; in this event, increasing the Frequency to 24 months is required until the situation is remedied as evidenced by satisfactory completion of two consecutive tests~~.

REFERENCES

1. 10 CFR 52.47(a)(2)(iv).
2. Regulatory Guide 1.183, July 2000.
3. 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors.”
4. Section 6.2.
5. Section 15.4.
6. Table 5.4-1.