HITACHI

GE Hitachi Nuclear Energy

James C. Kinsey
Vice President, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057

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Subject: Response to Portion of NRC Request for Additional Information Letter No. 100 - Containment Systems - RAI Numbers 6.2-158, 6.2-161, and 6.2-167

Enclosure 1 contains the GE Hitachi Nuclear Energy (GEH) response to the subject NRC RAIs transmitted via the Reference 1 letter.

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

Sincerely,

Kathy Sedney for

James C. Kinsey
Vice President, ESBWR Licensing

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NRO

Reference:

1. MFN 07-327, Letter from U.S. Nuclear Regulatory Commission to Robert Brown, *Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application*, May 30, 2007

Enclosure:

1. MFN 07-610 - Response to Portion of NRC Request for Additional Information Letter No. 100 - Related to ESBWR Design Certification Application - Containment Systems - RAI Numbers 6.2-158, 6.2-161, and 6.2-167

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GEH/San Jose (with enclosures)
	RE Brown	GEH/Wilmington (with enclosures)
	eDRF	RAIs 6.2-158 and 6.2-161: 0000-0075-3365
		RAI 6.2-167: 0000-0075-9372

Enclosure 1

MFN 07-610

Response to Portion of NRC Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application Containment Systems RAI Numbers 6.2-158, 6.2-161, and 6.2-167

NRC RAI 6.2-158:

Concerning DCD Tier 2, Rev. 3, Section 6.2.1.3:

During the ABWR review, the staff expressed concerns regarding the scaling loads used by GE for developing the load definition. To resolve this concern GE conducted ABWR-specific subscale (SS) and partial full-scale (FS) tests. The staff found this approach acceptable for the ABWR. However, it appears that GE has not demonstrated the applicability of the scaled test data to the ESBWR design.

GEH Response:

The ESBWR hydrodynamic load definitions and bases are described in the ESBWR Containment Load Definition Licensing Topical Report, NEDE-33261P (Reference 6.2-158-1). These include the loss-of-coolant accident (LOCA) Condensation Oscillation (CO) loads and the LOCA chugging (CH) loads.

CO loads occur at the main vent exit following a LOCA during periods of relatively high steam mass flux. The steam-liquid interface at the vent exit oscillates as the steam is condensed. The steam condensation produces periodic pressure loads on the suppression pool walls. The Advanced Boiling Water Reactor (ABWR) CO wall load was developed based on the subscale (SS) tests performed at the ABWR Horizontal Vent Test (HVT) facility.

CH loads occur at the main vent exit following a LOCA during periods of low mass flux. For a Design Basis Accident (DBA) LOCA, the chugging period will follow CO. The CH loads consist of a sharp pressure pulse followed by a damped oscillating pressure history. The ABWR CH wall load was established based on the partially full-scale (FS*) tests performed at the HVT facility

As described in Reference 6.2-158-1, the ESBWR load definitions are developed based on the corresponding ABWR loads. The applicability of the ABWR subscale and partial full-scale tests for ESBWR is demonstrated in Reference 6.2-158-1.

Section 4 of Reference 6.2-158-1 describes how the subscale tests are applicable to define the CO loads for ESBWR. Adjustments, as necessary are made to the ABWR CO load definition for ESBWR application. The adjustments are determined from a review of predicted thermal-hydraulic conditions during CO in the ESBWR, a review of the ESBWR and ABWR geometry, and a review of test data from the ABWR HVT subscale tests and tests from the Mark III Containment Pressure Suppression Test Facility tests.

Section 5 of Reference 6.2-158-1 describes how the subscale tests are applicable to define the CH loads for ESBWR. Adjustments to the ABWR definition for ESBWR application are determined from a review of expected conditions during CH in the ESBWR, comparisons of the ESBWR and ABWR geometry, and a review of the ABWR HVT FS* testing.

Therefore, the ESBWR containment loads report (Reference 6.2-158-1) demonstrates the applicability of both the CO and CH scaled test data to the ESBWR design.

Reference:

6.2-158-1: GE Hitachi Nuclear Energy, "ESBWR Containment Load Definition," NEDE-33261P, Class III (Proprietary), Revision 1, October 2007, and NEDO-33261, Class I (Non-proprietary), Revision 1, October 2007.

DCD Impact:

No DCD changes will be made in response to this RAI.

NRC RAI 6.2-161:

Concerning DCD Tier 2, Rev. 3, Section 6.2.1.3:

GE applies the Mark II hydrodynamic loads to the ESBWR design. The staff documented its evaluation of the definition of the Mark II design containment hydrodynamic load in NUREG-0808. In the evaluation of the pool swell phenomena (discussed in Section 2.1 of the NUREG report), the staff relied on comparisons with a substantial amount of data from tests conducted by both GE and Japan Atomic Energy Research Institute. These tests were directly applicable to the Mark II design. GE developed a computer program PSAM to be used as part of the Mark II hydrodynamic load evaluation program. The staff has reviewed the Mark II program and approved the methodology and PSAM in NUREG-0808. However, it did not find GE's methodology within PSAM acceptable. Rather, the staff based its acceptance on the favorable comparisons with the database. As a result, the use of the program for configurations other than those encompassed by the test data would not be accepted without further comparisons with applicable test data. Please, clarify the methodology used and provide a comparison with applicable test data as applied to the ESBWR.

GEH Response:

The Mark II program, as well as the Advanced Boiling Water Reactor (ABWR) certification, used the computer code PICSM to determine the pool swell hydrodynamic load. The PICSM code derives the key parameters used to define the impact loads above the suppression pool during pool swell.

The ABWR containment design is similar to the Mark III design in that it uses three rows of horizontal vents for blowdown to the suppression pool from the drywell. As part of the ABWR certification, GE justified the use of PICSM by comparing results from the code against the results from the Pressure Suppression Test Facility (PSTF), a Mark III-type test program. In the Safety Evaluation Report (SER) for ABWR (Reference 6.2-161-1, Section 6.2.1.6), the NRC accepted the use of PICSM to establish the pool swell response.

The ESBWR Containment Load Definition Licensing Topical Report (Reference 6.2-161-2) describes the methodology used to define the containment hydrodynamic loads including pool swell. That methodology is the same as that used for ABWR and reviewed by the NRC in the SER for ABWR (Reference 6.2-161-1). Reference 6.2-161-2 also describes the similarities between the ABWR and ESBWR containment designs and, thus, the applicability of using the ABWR methodology for ESBWR.

Therefore, the ESBWR containment design is very similar to the ABWR design, the methodology used to define the pool swell loads is the same as the methodology for ABWR, and comparison to applicable test data for this methodology was performed as part of the ABWR certification program.

References:

6.2-161-1: Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," NUREG-1503, July 1994.

6.2-161-2: GE Hitachi Nuclear Energy, "ESBWR Containment Load Definition," NEDE-33261P, Class III (Proprietary), Revision 1, October 2007, and NEDO-33261, Class I (Non-proprietary), Revision 1, October 2007.

DCD Impact:

No DCD changes will be made in response to this RAI.

NRC RAI 6.2-167:

In DCD, Tier 2, Revision 3, Section 6.2.3, the ESBWR reactor building (RB) should be subject to periodic functional testing. 10 CFR 50, Appendix J, Option A, states in IV.B that other structures of multiple barrier or subatmospheric containment (e.g. secondary containment for boiling water reactors and shield buildings for pressurized water reactors that enclose the entire primary reactor containment or portions there of) shall be subject to individual test in accordance with the procedure established in the technical specifications, or associated bases. Please provide information on the type of test that will be used to bound the RB leakage, the conditions under which the test would be run, the degree to which these conditions would reflect worst case accident conditions, the frequency of such test, and the establishment of a test criteria. This information may be coordinated with the response to RAI 6.2-165 regarding reactor building leakage.

GEH Response:

DCD Tier 2, Revision 4, Chapter 16 includes a Technical Specification Surveillance Requirement (SR) 3.6.3.1.4 that requires the following:

"Verify Reactor Building exfiltration rate within limits."

This requires a periodic functional test of the reactor building for leakage rate. This testing is discussed in the responses to RAIs 15.4-26 and 16.2-50. In summary, the reactor building air volume will be pressurized with a fan located outside the reactor building pressure boundary using an existing pipe penetration. The specific details related to performance of these tests are still being developed, including the conditions under which the test would be run, and the degree to which these conditions would reflect worst case accident conditions. The test frequency, as required by SR 3.6.3.1.4, is 60 months. The acceptance criterion is a reactor building leakage rate of less than 50% by weight per 24 hours. Also, as discussed in the response to RAI 15.4-26, DCD Tier 1 Table 2.16.5-2 (Inspection, Tests, Analyses and Acceptance Criteria for the Reactor Building) was revised in Revision 4 to add this test as a design commitment and to specify the acceptance criteria.

DCD Impact:

No DCD changes will be made in response to this RAI.