October 1, 2009

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-50
Wilmington, NC 28401

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 375 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Head:

By letter dated August 24, 2005, GE Hitachi Nuclear Energy (GEH) submitted an application for final design approval and standard design certification of the economic simplified boiling water reactor (ESBWR) standard plant design pursuant to 10 CFR Part 52. The U.S. Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application to enable the staff to reach a conclusion on the safety of the proposed design.

The NRC staff has identified that additional information is needed to continue portions of the review. The staff's request for additional information (RAI) is contained in the enclosure to this letter.

If you have any questions or comments concerning this matter, you may contact me at 301-415-3179 or Ilka.Berrios@nrc.gov or you may contact Amy Cubbage at 301-415-2875 or Amy.Cubbage@nrc.gov.

Sincerely,

/RA/

Ilka Berrios, Project Manager
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure:
Request for Additional Information

cc: See next page

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Dear Mr. Head:

By letter dated August 24, 2005, GE Hitachi Nuclear Energy (GEH) submitted an application for final design approval and standard design certification of the economic simplified boiling water reactor (ESBWR) standard plant design pursuant to 10 CFR Part 52. The U.S. Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application to enable the staff to reach a conclusion on the safety of the proposed design.

The NRC staff has identified that additional information is needed to continue portions of the review. The staff's request for additional information (RAI) is contained in the enclosure to this letter.

If you have any questions or comments concerning this matter, you may contact me at 301-415-3179 or Ilka.Berrios@nrc.gov or you may contact Amy Cubbage at 301-415-2875 or Amy.Cubbage@nrc.gov.

Sincerely,

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Ilka Berrios, Project Manager
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
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Enclosure:
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Distribution: See next page

ADAMS ACCESSION NO. ML092730598

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 375 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION DATED OCTOBER 1, 2009

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Requests for Additional Information (RAIs)
ESBWR Design Control Document (DCD), Revision 6

RAI Number	Reviewer	Question Summary	Full Text
RAI 6.2-140 S05	Wagage H/ Notafrancesco A	Provide updated RAI response, clarify implementation of PCC pool refill, and confirm vent fan discharge submergence and fan performance	In response to RAI 6.2-140, Supplement 4, GEH provided detailed results for 30-day containment analysis, which credited PCCS, PCCS pool refill, PARS, and *six* drywell gas recirculation (DGR) fans beyond three days. However, GEH later changed the design basis requiring operating only four of six DGR fans and provided selected results in ESBWR DCD Revision 6. A. The selected plots provided in ESBWR DCD Revision 6 are not sufficient for the staff to perform confirmatory analysis and complete the review. GEH should update RAI 6.2-140 response to include the four fan case, which represents the new design basis. B. GEH's response to RAI 6.2-140, Supplement 4, shows varying of the PCC pool refill rates to maximize the PCCS heat transfer. GEH should clarify how it will implement the controlled PCC pool refill rates during the operation of ESBWR. C. ESBWR DCD Revision 6 states that "The vent fan discharge line is > 0.22 m (9 in) and < 0.25 m (10 in) below the top of the drain pan lip." Confirm that the DGR fan discharge line would stay submerged during the 30 day analysis. D. ESBWR DCD Revision 6 Table 6.2-49 lists the minimum DGR fan performance requirements. Confirm that the DGR fan performance used in the containment analysis is bounded by the values listed in Table 6.2-49. E. ESBWR DCD Revision 6 Figure 6.2-14e1 shows curve of analytical pressure limit. a. Explain the regulatory significance of this curve. b. Explain how this curve would be used in ESBWR design.

RAI Number	Reviewer	Question Summary	Full Text
			c. Explain the relevance of this curve to ITAAC, if any. F. Update ESBWR DCD appropriately to include responses to parts B through D.
RAI 6.2-148 S02	Wagage H	Provide details of temperature predictions and testing to detect vacuum breaker leakage	The following refers to GEH's response to RAI 6.2-148, Supplement 1. A. Part A of the response states that "DCD Tier 1, Table 2.15.1-2, ITAAC 16b will be changed to a type test to detect bypass leakage from 0.3 cm^2 to 0.6 cm^2 (A/√K) using temperature sensors. Detecting leakage starting from 0.3 cm^2 (A/√K) assures the setpoint calculation will have margin to the 0.6 cm^2 (A/√K) analytical limit to close a VBIV." Describe test conditions and how the tests will be performed, and clarify ITAAC to state that the design basis accident conditions cover 30 days following a loss-of-coolant accident. B. Part E of the response notes that an 1-D vacuum breaker component was modeled in TRACG. Provide the initial and boundary conditions of the model and the TRACG input file for staff review. C. Table 1 in the response provides predicted steady state temperature differential between two selected locations and the wetwell air space. a. Considering transient conditions in the drywell and wetwell, explain the validity of steady state temperature preditions to make setpoints for detecting vacuum breaker leakage and closing the vacuum breaker isolation valves. b. Following its response GEH stated during a telephone call that the boundary conditions used for calculation were from drywell and wetwell pressures and temperatures at 50 seconds after blowdown for the bounding case of the main steam line break accident in the containment. A significant temperature difference exists between the drywell and wetwell at the selected time, and

RAI Number	Reviewer	Question Summary	Full Text
			therefore, resulting in significant temperature differences between the selected locations and the wetwell air space. However, temperature difference between the drywell and wetwell diminishes with time. When vacuum breakers open after 72 hours following a LOCA, considering accuracy of temperature sensors, confirm that the predicted temperature difference between a selected location and the wetwell can be reasonably predicted to detect vacuum breaker leakage.
9.3-45	Hinson C	Verify the accuracy of the following DCD changes made in Rev. 6 of the DCD: a) deletion of reference to PSS in Table 1 A-1, b) deletion of reference to conformance with 10 CFR 50.34(f)(2)(viii) in Section 9.3.2.	In Revision 6 to the DCD, several changes were made with respect to the use of the Process Sampling System (PSS) to obtain post accident samples. The staff has the following concerns with respect to some of these changes: a) In Revision 5 to the DCD, Table 1 A-1 (section related to conformance with 10 CFR 50.34(f)(2)(viii)) stated that the ESBWR Containment Monitoring System (CMS) and the PSS provide contingency plans for obtaining and analyzing highly radioactive reactor coolant, suppression pool, and containment atmospheric samples. In Revision 6 to the DCD, references to Section 9.3.2 and the PSS were deleted while reference to the CMS was retained in this table. Since reference to the use of the PSS was deleted, verify that the various post-accident actions listed in this section of this table (i. e., obtaining and analyzing highly radioactive reactor coolant, suppression pool, and containment atmospheric samples) can be obtained using the CMS only. If some of these post-accident samples can only be obtained by way of the PSS, then references to the PSS and Section 9.3.2 should be reinserted in this section of Table 1 A-1. b) In DCD Section 9.3.2, reference to 10 CFR 50.34(f)(2)(viii) was deleted and a statement added stating that conformance to 10 CFR 50.34(f)(2)(viii) is not required. 10 CFR 50.34(f)(2)(viii) states that a capability must be provided to promptly obtain and analyze samples. Even though GEH states that the ESBWR design will not have a formal PASS, there still must be contingency plans to obtain and analyze post-accident samples. Therefore, GEH must still comply with the requirements of 10 CFR 50.34(f)(2)(viii) and reference to this requirement should be reinserted into Section 9.3.2.
12.3-14	Hinson C	Modify DCD Section 12.3.2.2 to include a description of the functions, use, and technical justification	Regulatory Guide 1.206 (Section C.I.12.3.2) states that the applicant should provide information regarding the shielding for each radiation source identified in Chapter 11 and Section 12.2, including the assumptions, codes, and techniques used in the shielding calculations. Section 12.3.2.2.2 of the ESBWR DCD contains a description of several of the shielding codes (QADF, GGG, DORT)

RAI Number	Reviewer	Question Summary	Full Text
		for use of the following computer codes- EMIR, PANDORA, NISEIS, and MCNP that are referenced in E.A. shielding calculational packages. Include a listing of these codes in DCD Table 12.3-1.	used by GEH in performing their shielding design. In addition, Table 12.3-1 lists these shielding codes and others identified in other sections of the DCD. During the recent staff review of several shielding calculational packages provided to the staff by GEH in preparation for the staff's September 14-18, 2009 QA inspection at GEH offices, the staff noted that GEH's contractor, Empresarios Agrupados (E.A.), had utilized the following shielding codes, EMIR (2.0) and PANDORA (4.0) to perform the shielding calculations to determine the dose rates from the radwaste tanks and other components that are part of the Solid Waste Management System (SWMS). In addition, one of the shielding packages reviewed makes reference to two other codes, NISEIS and MCNP, as being referenced in the supplementary documents listed as part of the shielding calculational package. Even though these codes were apparently used by GEH in the determination of dose rates (and subsequent radiation zone designations) from SWMS plant components, no description of any of these codes is mentioned anywhere in Chapter 12 of the ESBWR DCD. It is the staff's position that if GEH used these codes as part of the shielding analysis for radiation sources identified in Chapters 11 and 12 of the ESBWR DCD, then GEH should modify Section 12.3.2.2.2 of the ESBWR DCD to a. include reference to the use of these shielding codes, b. include a description of the function of each of these codes, and c. provide a general listing of what plant systems/components were modeled using each of these shielding codes. The staff noted that Section 12.3.2.2.2 of the DCD currently describes the use of several codes (QADF, GGG, and DORT) which have been approved by the NRC and which are available through the Radiation Safety Information Computational Center (RSICC). Based on the staff's review of the above mentioned E.A. shield calculational packages, several of the codes mentioned in these packages appear to be similar in function to the shield codes referenced in the DCD.

RAI Number	Reviewer	Question Summary	Full Text
			d. Provide a comparison of the capabilities of the E.A. identified codes with the shielding codes currently referenced in the DCD. e. Provide a justification in the DCD of why the above mentioned shielding codes in the E.A. calculational packages represent acceptable alternative shielding codes to the comparable codes which have been approved by the NRC and which are available through the RSICC. f. Include a listing of these E.A. referenced shielding codes in DCD Table 12.3-1.
12.3-15	Hinson C	Describe potential high radiation doses from inadvertent reactor cavity draindown	Background GDC 61 requires that the fuel storage and transfer system, in addition to any other system which may contain radioactivity, be designed to ensure adequate safety and shielding, as well as designed to prevent the release of radioactive material during normal and accident conditions. Generic Safety Issue 137 (GSI-137) “Refueling Cavity Seal Failure” was initiated to consider a Reactor Cavity Seal Ring failure as an initiating event for a Spent Fuel Pool accident sequence (see NRC Bulletin 84-03). GSI-137 noted the following possible consequences to a reactor cavity seal ring failure; (1) high radiation levels in the containment due to uncovering of spent fuel during transfer, (2) radioactive material release in the containment building due to rupture of fuel pins (by self-heating after uncovering), (3) high radiation levels in the spent fuel building due to uncovering of stored spent fuel, and (4) radioactive material release outside the containment building due to rupture of fuel pins in the storage pool. Mitigative measures to prevent the recurrence of this event and the possible resulting drainage of the spent fuel pool in the spent fuel building include the installation of permanent reactor cavity seal rings and the installation of coffer dams between the refueling cavity and the spent fuel pool to prevent draining the spent fuel pool. In their response to NRC Bulletin 84-03, "Refueling Cavity Water Seal," a number of licensees indicated that if a large seal leak occurred, the refueling cavity could drain rapidly and the procedures and makeup capability for refilling the refueling cavity would be insufficient to prevent the potential uncovering of a

RAI Number	Reviewer	Question Summary	Full Text
			fuel assembly in transit. In anticipation of such an event, the licensees developed an abnormal occurrence procedure for safely storing a fuel assembly under water if one was positioned above the vessel flange level in the event of a loss of the refueling cavity water. Question GSI 137 addresses rapid reactor cavity drain down due to cavity seal failure and the possibility of associated spent fuel pool drain down and the uncovering of stored spent fuel. GEH should address the following issues as they relate to high radiation doses and/or releases of radioactive material caused by inadvertent draindown of the reactor cavity: a. In the event that the reactor cavity underwent a rapid inadvertent drain down during refueling operations when a spent fuel assembly was positioned above the reactor vessel flange level, describe the location(s) (such as the buffer pool or the reactor vessel) where this spent fuel assembly could be safely lowered to minimize the potential for potential high radiation levels, fuel clad damage, and the subsequent release of radioactive material. For each of the possible locations available for fuel laydown, state the expected resultant water level over the top of the active portion of the fuel assembly following cavity draindown. For those locations where the water level at the lowest point following the drain down event is less than 10 feet above the fuel, provide the estimated resultant dose rates, and the associated methods, models and assumptions used to calculate these dose rates. b. If the volume of water in the safe storage area is less than the amount to needed to ensure complete coverage of the fuel bundle for the 30 minutes allotted for ensuring containment closure, then justify the continued use of the Fuel Drop Accident Analysis methodology described in RG 1.183 Appendix B. c. Describe and provide the estimated dose rates, from any other irradiated components, that may be exposed during an inadvertent refueling cavity drain down event

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