

September 15, 2009

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-18
Wilmington, NC 28401

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 371 RELATED TO
ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Head:

By letter dated August 24, 2005, GE Hitachi Nuclear Energy 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 301-415-6715 or Bruce.Bavol@nrc.gov or you may contact Amy Cabbage at 301-415-2875 or Amy.Cabbage@nrc.gov.

Sincerely,

/RA/

Bruce M. Bavol, 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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Distribution: See next page

ADAMS ACCESSION NO. ML092540353

NRO-002

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DATE	09/15/2009	09/15/2009

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 371 RELATED TO
ESBWR DESIGN CERTIFICATION APPLICATION DATED
SEPTEMBER 15, 2009

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

RAI Number	Reviewer	RAI Summary	RAI Text
4.4-23, Supplement 4 (MFN 09-556 dated August 19, 2009)	Gilmer J	Demonstrate that the 10CFR50.46 PCT limit will not be exceeded following a LOCA or MSLB when partial or complete blockage of some fuel bundles occurs due to debris.	<p>In the response to RAI 4.4-23, Supplement 3 (MFN 09-556), it is stated that the LOCA water level is always above the top of the active fuel, and, in the case of a Main Steam Line Break, will be maintained at or above 7 m (23 ft) after around 1600 seconds. Following a LOCA or Main Steam Line Break, the liquid in all fuel channels will flash to vapor during the initial rapid depressurization. The liquid will be gradually replaced by GDCS flow once the squib valves are opened. The staff believes it is possible to retain or form a vapor bubble within the fuel channels which are partially or totally blocked due to debris buildup at the inlet orifice and debris filter screen. This is most likely early in the transient, when the up-flow due to boil off of water present in the channels (prior to blockage of the inlet flow path) exceeds the gravity-driven down-flow from the static water head above the channels. Later in the transient, the GDCS pool water flow from the top of the blocked channels will offset the vapor up-flow, and water will be restored.</p> <p>Provide a calculation for the limiting break size, type, and location that demonstrates that the 10CFR50.46(b)(1) limit for peak clad temperature, 2200°F, will not be exceeded, or, if this cannot be shown for a limited number of bundles, demonstrate that the fuel failure acceptance criterion (upon which the radiological source term is based) is not exceeded. The calculation should be similar to that presented by the BWR Owners Group for BWR 2 through 6, which was presented at the July 23, 2009 meeting at NRC Headquarters.</p> <p>If the TRACG code is used to perform the analysis, provide the basis for code qualification at low pressure (< 700 psia) for the GEXL critical power correlation and also the TRACG countercurrent flow model.</p> <p>Also, provide an explanation for the significance of the 1600 seconds following a Main Steam line break for the 7 m (23 ft) of water above the top of active fuel (i.e., a timeline for reflood to this level above the core).</p>

Enclosure

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(Revised 09/09/2009)

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