HITACHI

GE Hitachi Nuclear Energy

Richard E. Kingston
Vice President, ESBWR Licensing

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

T 910 675 6192
F 910 362 6192
rick.kingston@ge.com

MFN 08-579

Docket No. 52-010

July 29, 2008

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application – Design of Structures, Components, Equipment, and Systems - RAI Number 3.9-81**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to a portion of the U.S. Nuclear Regulatory Commission Request for Additional Information (RAI) sent by NRC Letter 67, Dated October 10, 2006 (Reference 1). The GEH response to RAI Number 3.9-81 is addressed in Enclosure 1.

If you have any questions about the information provided here, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 06-378, Letter from the U.S. Nuclear Regulatory Commission to David H. Hinds, Manager ESBWR, Request for Additional Information Letter No. 67, Related To ESBWR Design Certification Application, Dated October 10, 2006

Enclosure:

1. Response to Portion of NRC Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application – Design of Structures, Components, Equipment, and Systems - RAI Number 3.9-81

cc:	AE Cubbage	USNRC (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	eDRF	0000-0077-9032 Revision 1

Enclosure 1

MFN 08-579

Response to Portion of NRC Request for

Additional Information Letter No. 67

Related to ESBWR Design Certification Application

Design of Structures, Components, Equipment, and Systems

RAI Number 3.9-81

NRC RAI 3.9-81

It is stated in DCD Tier 2, Section 3.9.2.5 that the reactor internal pressure differentials due to an assumed break in main steam or feedwater line are determined by analysis as described in DCD Tier 2, Section 3.9.5.3. In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces during an accident, a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a comprehensive vertical dynamic model of the reactor pressure vessel (RPV) and internals. Provide a summary of the analytical results that demonstrate no significant dynamic amplification of the loads on the reactor internals core support structures as a result of the postulated break in main steam or feedwater lines.

GE Response

The only reactor internals vertical core support structure, in the GE ESBWR nuclear reactor, is the Control Rod Guide Tubes/Control Rod Drive Housings (CRGTs/CRDHs). An assemblage of axial beam elements is employed to represent the CRGTs/CRDHs fuel vertical inertia load path in the vertical primary structure model. No vertical natural frequency, below 105.4 Hz, was obtained for the CRGTs/CRDHs load path in the vertical primary structure model eigen analysis. Furthermore, the blowdown loads associated with the main steam and feedwater line breaks do not excite the CRGTs/CRDHs fuel core vertical inertia load path. Consequently, there can be no dynamic amplification of the blowdown loads through the CRGTs/CRDHs load path.

DCD Impact

RAI 3.9-81 was issued prior to submittal of Rev. 5 of the DCD. DCD Tier 2, Subsection 3.9.2.5, Rev. 5 has been revised to address this issue. Loading conditions for RPV internals are described in DCD Tier 2, Rev. 5, Subsection 3.9.5.3.