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MFN 06-368
Supplement 2

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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. 56 Related to ESBWR Design Certification Application – DCD Chapter 17 - RAI Number 17.4-1 S01**

Enclosure 1 contains GEH's response to the subject NRC RAI transmitted via e-mail on February 27, 2007. GEH's original response was provided in the Reference 1 letter.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 06-368, Letter from David H. Hinds to U.S. Nuclear Regulatory Commission , *Response to NRC Request for Additional Information Letter No. 56 – DCD Chapter 17 – RAI Numbers 17.1-1, 17.2-1, and 17.4-1 through 17.4-12*, dated October 6, 2006

Enclosure:

1. MFN 06-368, Supplement 2 - Response to Portion of NRC Request for Additional Information Letter No. 56 Related to ESBWR Design Certification Application – DCD Chapter 17 - RAI Number 17.4-1 S01

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH (with enclosure)
	RE Brown	GEH (with enclosure)
	DH Hinds	GEH (with enclosure)
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Enclosure 1

MFN 06-368

Supplement 2

Response to Portion of NRC Request for

Additional Information Letter No. 56

Related to ESBWR Design Certification Application

DCD Chapter 17 - RAI Number 17.4-1 S01

For historical purposes, the original text of RAI 17.4-1 and the GE response is included. The RAI 17.4-1 response does not include the attachments or DCD mark-up previously transmitted.

NRC RAI 17.4-1

The Advanced Boiling Water Reactor (ABWR) Standard Safety Analysis Report (SSAR) included Table 19K-1, "ABWR SSCs of Greatest Importance for CDF - Level 1 Analysis," which listed the risk significant SSCs along with probabilistic risk assessment (PRA) importance measure thresholds including risk rankings values for risk significant SSCs within the scope of RAP. This information is not provided in ESBWR DCD Tier 2, Sections 17.4. Some of this information is provided in NEDC-33201P, Table 19-1, "ESBWR SSCs of Greatest Importance for CDF and Level I Analysis," and Table 19-2,"ESBWR Initiating Event Contribution to CDF, Level 1 Analysis."

A reference should be added to DCD Tier 2, Section 17.4, "Reliability Assurance Program During Design Phase," to the list of risk significant systems, structures, and components (SSCs) identified in the NEDC-33201P, Tables 19-1 and 19-2. The applicant should also add references to identify risk significant SSCs within the scope of D-RAP identified from PRA Level-I analysis for external events, PRA Level-II analysis, engineering judgment and operating experience supporting risk insights, and the expert panel process. The applicant should ensure that the list is all-inclusive of SSCs that have been identified to be within the scope of D-RAP.

GE Response

A reference will be added to the next revision of DCD Tier 2, Section 17.4, "Reliability Assurance Program During Design Phase," to the list of risk significant systems, structures, and components (SSCs) identified in the DCD Tier 2, Section 19, Table 19.1-3. This list is based on preliminary Level-1 PRA.

The task of identifying a comprehensive list of risk significant SSCs within the scope of D-RAP will be performed in a later phase of development for the D-RAP. The list will be all-inclusive of SSCs that have been identified to be within the scope of D-RAP. In addition to the preliminary Level-I internal events insights, the list will include SSCs identified from PRA Level-I analysis for external events, PRA Level-II analysis, engineering judgment and operating experience supporting risk insights, and the expert panel process.

The process of developing and maintaining the list of risk-significant SSCs is described in the Reliability Assurance Program Plan, NEDO-33289. The list of risk significant SSCs will be controlled as an issued design specification, 26A7107 ESBWR Risk Significant SSCs.

DCD Impact

DCD Tier 2, Section 17.4 will be revised in the next update as noted in Enclosure 2.

NRC RAI 17.4-1 S01

In response to RAI 17.4-1, GE stated that the task of identifying a comprehensive list of risk significant SSCs within the scope of D-RAP will be performed at a later phase of development of the D-RAP. The process of developing and maintaining the list of risk significant SSCs is described in NEDO-33289 and design specification 26A7107, "ESBWR Risk Significant SSCs."

GE plans to complete the list of risk significant SSCs for the ESBWR design certification application in the near future. As described above, the list of risk significant SSCs will be maintained in design specification 26A7107. The DC applicant must provide design specification 26A7107 to the NRC staff and reference this document in DCD Section 17.4 so that the NRC staff can complete its review of the ESBWR D-RAP. RAI 17.4-1 is being trakced as a preliminary open item.

GEH Response

The list of risk significant SSCs for the ESBWR design certification application is maintained in GEH Licensing Topical Report NEDO-33411, "Risk Significance of Structures, Systems and Components For the Design Phase of the ESBWR." This report was prepared as a topical report to be consistent with GEH reporting formats and it replaces design specification 26A7107 entirely. NEDO-33411, Revision 0 has been submitted to the NRC (Reference MFN 08-277).

DCD Impact

DCD Tier 2, Section 17.4 will be revised as noted in the attached markup. Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

LTR NEDO-33411, Rev 0 has been created as described above.

17.4.6 SSC Identification/Prioritization

A list of risk-significant SSCs is developed and controlled as a ~~design specification document~~topical report (Reference 17.4-7). The preliminary list is based on the results of the generic PRA. The list is updated when the plant-specific PRA is developed. At this point, a blended approach is used for identifying and prioritizing risk significant SSCs. This approach combines the various PRA analytical results with operating experience and an expert panel process to develop a comprehensive risk analysis.

The level 1 PRA is used to evaluate accident sequences from initiating events and failures of safety functions that lead to core damage. An assessment is performed for operating and shutdown conditions. The external events analysis considers events whose cause is external to systems associated with normal plant operations, including internal flooding, fire, high winds, and seismic events. The seismic events are analyzed using a seismic margins approach that provides qualitative conclusions on the ability of ESBWR SSCs to cope with seismic events. The other external events are quantified using the level 1 PRA.

Level 1 basic events representing component failures are identified as risk-significant if their importance values for Risk Achievement Worth (RAW) are greater than or equal to 5.0, or Fussell-Vesely Importance are greater than or equal to 0.01.

Level 2 risk significance is determined by identifying the dominant contributors to severe accidents and offsite release of fission products. This qualitative analysis, which is performed by the expert panel, includes the evaluation of severe accident phenomena and fission product source terms, and containment integrity strategies including pressure suppression, decay heat removal, and hydrogen generation.

SSC functions relied upon under power-operating and shutdown conditions to meet the NRC's safety goal guidelines of a Core Damage Frequency (CDF) of less than 1.0E-4 per reactor year and Large Release Frequency of less than 1.0E-6 per reactor year are risk-significant. SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents are also risk-significant.

Operating experience identifies previous failures of components in similar applications, and also reveals situations where inappropriate human actions have led to functional failures of SSCs. The expert panel assesses component operating history and industry operating experience when it can be applied to assessing risk significance.

Safety-related SSCs are controlled by plant Technical Specifications. If a nonsafety-related SSC is shown through operating experience or PRA to be significant to public health and safety, then it should be controlled by Technical Specifications. In this case, "significant" equates to an SSC that is required to meet the NRC Safety Goals. If it is determined that an SSC is risk significant, but is not required for meeting the NRC Safety Goals, then performance controls should be implemented through the RAP. If the SSC is not significant, then normal controls would be implemented through the site Maintenance Rule and corrective action programs.

17.4.7 Design Considerations

The reliability of risk-significant SSCs, which are identified by the PRA and other sources, are evaluated at the detailed design stage by appropriate design reviews and reliability analyses. The procedure for design change control defines the process for evaluating design changes in

17.4.14 References

17.4-1 GE Energy Nuclear, "Reliability Assurance Program Plan", NEDO-33289, ~~October 2006~~Revision 1, December 2007.

17.4-2 US Nuclear Regulatory Commission, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)", SECY-95-132, May 1995.

17.4-3 US Nuclear Regulatory Commission, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Regulatory Guide 1.160, March 1997.

17.4-4 US Nuclear Regulatory Commission, "Assessing and Managing the Risk Before Maintenance at Nuclear Power Plants," Regulatory Guide 1.182, May 2000.

17.4-5 NEI, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01 April 1996.

17.4-6 NEI, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision to Section 11 of NUMARC 93-01 February 22, 2000.

17.4-7 GE Hitachi Nuclear Energy, "Risk Significance of Structures, Systems and Components for the Design Phase of the ESBWR," NEDO-33411, Revision 0, March 2008.