

January 14, 2008

APPLICANT: GE-Hitachi Nuclear Energy Americas LLC (GEH)

PROJECT: Economic Simplified Boiling Water Reactor (ESBWR)

SUBJECT: SUMMARY OF ESBWR PROBABILISTIC RISK ASSESSMENT AUDIT
CONDUCTED ON OCTOBER 25 and 26, 2007

The U.S. Nuclear Regulatory Commission (NRC) conducted an ESBWR Probabilistic Risk Assessment (PRA) audit on October 25 and 26, 2007, at the GE-Hitachi Nuclear Energy facility in Wilmington, North Carolina. The NRC staff conducted the audit with representatives of GEH. Attendees are identified in Enclosure 1.

The NRC staff conducted the audit as a follow-up to the July 18 and 19, 2007, audit as part of the agency's review process for ESBWR design certification. The audit included review of GEH's responses to requests for additional information (RAIs) associated with staff's review of the ESBWR Design Control Document (DCD), GEH's topical report (TR) NEDO-33201, Revision 2, "ESBWR Probabilistic Risk Assessment," TR NEDO-33386, "Flood Zoning Report," and review of various technical assumptions and analyses pertaining to risk assessment.

The audit focused on PRA Level 1 internal events results; PRA quality; Level 2 PRA and severe accidents; internal flooding modeling at power; internal fire analysis; shutdown analyses and other technical issues. The audit agenda is provided as Enclosure 2. The audit involved specific technical discussions between the staff and GEH representatives which are described as follows:

Level 1 Internal Events (at power) Results

The staff requested that the applicant summarize the reasons for the decrease in the overall core damage frequency from Revision 1 of the PRA to Revision 2, including any changes to initiating event and basic event data. The applicant explained that changes to data were not a major factor because some key data values increased while others decreased. The applicant stated that the main reason for the decrease in core damage frequency was an ESBWR design change that is reflected in Revision 2 of the PRA. Following submittal of Revision 1 of the PRA, the ESBWR design was modified such that a substantially larger amount of water is injected into the reactor vessel when the isolation condenser system (ICS) is actuated. This change improved the effectiveness of the ICS as a means of high pressure injection. Loss of preferred power and loss of feedwater transient sequences which, prior to the design change, required high pressure injection of 3 of 4 isolation condensers and 2 of 2 control rod drive pumps to restore vessel water levels are now mitigated at high pressure by successful performance of the ICS without need for CRD pumps. Prior to the design change, failure of a CRD pump necessitated depressurization and use of low pressure systems to restore level. Consequently, available isolation condensers (which function at high pressure) could not be used for decay heat removal. This change resulted in the elimination of a number of core damage sequences

associated with failure of a single CRD pump. The staff is currently evaluating the information provided by the applicant.

At the staff's request, the applicant clarified the way in which the sensitivity study associated with safety relief valve common cause grouping and the passive component demand study were performed. The applicant explained the reasons for performing these particular studies and some of the methods used to perform the studies.

PRA Quality

In a previous teleconference with the applicant, the staff requested that they clarify statements in Revision 3 of the Design Control Document (DCD) regarding application of the ASME Standard for Level 1 PRA to the ESBWR PRA. In the October 25, 2007, meeting, the applicant described the extent to which requirements in the standard were used to guide development of the PRA and how certain requirements were found to be not applicable to a design PRA. The applicant also described their program for systematically assessing the extent to which the current version of the PRA meets the requirements of the Standard and reconciling instances where the PRA does not comply with the Standard. Following this discussion with the applicant, the staff examined the quality assurance procedures used in the development of the ESBWR PRA. The staff was able to confirm from this examination that quality control of the PRA process was performed in accordance with procedures that comply with Appendix B of 10 CFR Part 50. The staff indicated that it would issue a formal request for information to ensure that the information provided at the meeting is documented in a form that can be referenced in the staff's safety evaluation report.

Level 2 PRA and Severe Accidents

Discussions were held on the following topic areas:

- Changes to Section 8 of the PRA
 - GEH's rearrangement of the containment event trees in Section 8 of the PRA is more consistent with the Level 1 sequences.
 - The "physically unreasonable" approach is more consistent.
 - The tracking of accident sequences into release categories was improved
 - Discussion of the results of MAAP runs in the report has been improved.
- The consequence analysis results reported in Section 10 of the PRA were carried out using an updated version of MACCS2. GEH stated that the results bound potential ESBWR site results
- Section 11, on uncertainty analysis, was greatly expanded
- Section 21 on severe accidents has been revised again. It does not include comments from the peer reviewers, although a separate stand-alone document does. The BiMAC test report was not included.

- The BiMAC test report has not yet been released.
- Severe Accident Management was included as part of the human factors evaluation in Chapter 18 of the DCD. GEH is working with the COL applicants to put together EPG/SAG guidelines. A plan outline was presented in a topical report referenced in Chapter 18.
- Several outstanding RAIs were discussed.
- A number of potential new RAIs were discussed. Clarification was provided on some RAIs at the meeting and the remainder to be sent as supplemental RAIs.

Internal Flooding Modeling at Power:

GEH explained the method for analyzing the flooding scenario at power. For postulated flood events occurring at power, the general transient initiating event category and associated accident sequence logic is used to model the accident sequence progression. The calculated flood initiator frequency and associated equipment impacts are propagated through the general transient Level 1 internal events accident sequence logic for the flood scenario. The staff and GEH discussed a few of the top cutsets in the cutest reports.

GEH explained that not all flooding zones (e.g., Service Water Building) are presented in the flood zoning report (NEDE-33386). This is due to the fact that the ESBWR DCD only addresses the generic plant configuration. The detailed plant configuration will be based on the specific plant design during the COL application phase. However, the flooding damage due to generic plant configuration is evaluated and the results can be used by the COL applicant if plant specific results are bounded by the DCD.

GEH discussed the guidelines to model the flood propagation:

- Once flood reaches the stairwell, it will flow to the lowest elevation of the stairwell and will not flood intermediate floors.
- Because of the large opening in the floors of the turbine building, it is assumed that flooding will propagate through to the basement elevation without flooding intermediate floors.
- Watertight doors do not fail.

The staff found that the guidelines are reasonable.

Internal Fire Analysis

GEH discussed that the probabilistic fire analysis is performed with reasonable assumptions since the specifics of cable routings, ignition sources, or target locations in each zone of the plant are still in the design phase. Because of this limitation, a simplified, conservative, and bounding approach is used in this analysis.

GEH stated that there are iterative discussions among the component engineer, HVAC engineer, I&C engineer, fire protection engineer, and PRA analyst to design the plant in an optimal fashion from the fire protection point of view.

The staff found that this approach is reasonable.

The staff considered the audit to be a valuable exchange of PRA-related information and assisted the staff in its review of the ESBWR PRA.

Please direct any inquiries concerning this audit to me at 301-415-5787 or rdf@nrc.gov.

/RA/

Rocky D. Foster, Project Manager
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Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-10

Enclosures:
As stated

cc w/encl: See next page

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DATE	11/20/2007	11/20/2007	11/27/2007	1/14/2008

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Meeting Summary for ESBWR PRA Audit by Rocky D. Foster dated January 14, 2008

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ESBWR Probabilistic Assessment Audit

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October 25 and 26, 2007

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Ed Fuller	NRC/NRO/DSRA
John Lai	NRC/NRO/DSRA
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SUMMARY OF ESBWR
PROBABILISTIC RISK ASSESSMENT
AUDIT CONDUCTED
OCTOBER 25 and 26, 2007

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