December 8, 2005

Mr. David Hinds, Manager, ESBWR
General Electric Company
P.O. Box 780, M/C L60
Wilmington, NC 28402-0780

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 3 FOR THE ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Hinds:

By letter dated August 24, 2005, General Electric Company (GE) submitted an application for final design approval and standard design certification of the economic simplified boiling water reactor (ESBWR) standard plant design. The application included a Design Control Document (DCD) and a Probabilistic Risk Assessment (PRA) for the ESBWR. The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application. The NRC staff has determined that additional information is needed to continue portions of the review.

Enclosure 1 contains a request for additional information (RAI) regarding information contained in the PRA and DCD Chapter 19, PRA & Severe Accident. The RAIs were discussed with you and your staff during a telephone call on September 15, 2005, a meeting on September 29, 2005, and a telephone call on November 9, 2005. During the November 9, 2005, telephone call you agreed to provide a response to the requested information by December 30, 2005, and to include resulting changes to the PRA in NEDO-33201, “ESBWR Probabilistic Risk Assessment,” Revision 1, which is scheduled to be submitted on January 6, 2005. You also indicated that the responses to RAI 19.1.0-1 and RAI 19.1.0-2(b) may be a partial response, due to the technical complexity of the information requested. If so, your initial RAI response will provide a commitment for the completion date for these two questions.

If you have any questions or comments concerning this matter, you may contact me at (301) 415-2863 or lwr@nrc.gov,or you may contact Amy Cubbage at (301) 415-2875 or aec@nrc.gov.

Sincerely,

/RA A. Cubbage for:/

Lawrence Rossbach, Project Manager
New Reactor Licensing Branch
Division of New Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 52-010

Enclosure: As stated

cc: See next page

Request for Additional Information - ESBWR PRA and Chapter 19 of the Design Control Document

RAI Number	Reviewer	Summary	Full Text
19.0.0-1	R. Palla	Provide peer review results.	Peer review is an essential part of the ROAAM methodology used to support the assessment of direct containment heating, steam explosions, and core concrete interactions for ESBWR. General Electric (GE) indicates that the results of an independent review are provided in NEDC-33201 (see DCD p. 19.3-5 and PRA p. 21.2-3), but this information appears to have been omitted. Please provide this information. The expert's reports and author's responses are essential to establishing the credibility of the failure probability values assigned for these phenomena (typically 0.001 or 0.01). In the absence of this documentation, considerably greater staff review resources and time will be required.
19.2.3-1	R. Palla	Provide an equipment survivability assessment.	Other than mentioning the equipment survivability in the context of SECY-93-087 (DCD p. 19.1-2), GE does not appear to have included any assessment of equipment survivability - in either the DCD or the PRA. Please provide an equipment survivability assessment.
19.2.4-1	R. Palla	Provide accident management program discussion or commitment.	Provide a discussion or commitment (combined operating license action item) regarding the accident management program under which guidance and training would be provided on the use of such features as containment venting, drywell sprays, and AC-independent fire pumps for isolation condenser make-up.

19.4.0-1	R. Palla	Provide more rigorous SAMDA assessment of design alternatives or enhancements.	The evaluation of Severe Accident Mitigation Design Alternatives (DCD Section 19.4.9) does not include the identification and evaluation of specific design alternatives or enhancements. Instead, it provides an estimate of the dollar value of the residual risk for the ESBWR, and rationale as to why plant improvements that provide a measurable reduction in severe accident risk are not likely to have implementation costs less than this value. Although this rationale may be technically sound, the evaluation does not provide the "hard look" at design alternatives required by the National Environmental Policy Act. Provide a more rigorous assessment, considering the risk profile for the ESBWR, potential improvements to further reduce the dominant risk contributors, and the costs and benefits of these improvements.
19.0.0-2	R. Palla	Include the contribution from all accident classes.	In determining the large release frequency (LRF), conditional containment failure probability (CCFP), and total population dose for the ESBWR, GE neglected the contribution of Class II, IV, and V sequences. Class II sequences, which contribute about 8 percent of the core damage frequency (CDF), were neglected because these sequences do not result in core damage until after 72 hours and are recoverable with manual actions. The other sequence classes were neglected based on their small contribution to CDF. Although releases for Class II sequences would not generally be considered "early," they could still result in substantial consequences and impact the risk metrics (e.g., inclusion of all accident sequence classes could increase the CCFP from about 3 percent to 12 percent). Accordingly, the contribution from all accident classes needs to be included when characterizing the overall containment performance and risk for severe accidents.

19.0.0-3	R. Palla	Address lower drywell flooding issues.	Flooding of the lower drywell (LDW) prior to reactor vessel breach is a key determinant of the probability of containment failure due to ex-vessel steam explosions. No description is provided of the flow paths into the LDW, and the basis for GE's estimated probabilities of various pre-existing water levels in the LDW. Also, inconsistent sets of probability values are provided for the LDW water heights (i.e., PRA p. 21.4-5 indicates the likelihood of a high, medium, and low water level is 5 percent, 59 percent, and 36 percent, respectively, whereas PRA p. 8.3-4 indicates values of 0.9 percent, 0.1 percent, and 99 percent). This calls into question GE's quality assurance of the document, and could substantially impact PRA results. Please address these LDW flooding issues.
19.0.0-4	R. Palla	Address BiMAC issues.	The Basemat Internal Melt Arrest and Coolability (BiMAC) device appears to have been developed only to a conceptual level. For example, based on discussions in DCD Section 19.3.5 and PRA Section 21.5, the top plate, refractory plate, and grating that will cover the BiMAC have not been finalized, the BiMAC sacrificial material and its thickness have not yet been specified, the positioning and dimensioning of the cooling jacket and the angle of inclination have not been established, and the testing of BiMAC needed for confirmation and optimization has not yet been performed. Also, BiMAC actuation relies on squib valves operated from thermocouples embedded in the drywell floor, or on additional valves that would be passively actuated through melting of eutectic alloys exposed to high temperatures in the LDW. The design details of the thermocouple actuation system and eutectic-based valves, as well as the process for establishing the assumed 1E-3 failure frequency for the BiMAC system are not provided. In the absence of further design information and experimental validation, the credit taken in the Level 2 PRA for BiMAC arresting core melt progression (assumed to be 99 percent effective) is questionable. Without credit for BiMAC, it appears that events that proceed to reactor vessel breach will result in either containment venting or over-pressure failure at about 15 hours, thereby substantially impacting the results of the Level 2 and 3 PRA analysis. Please address these BiMAC issues.

19.0.0-5	R. Palla	Address protection of the LDW sumps by the BiMAC cooling jacket and address the corium splash shield.	The protection of the LDW sumps by the BiMAC cooling jacket is only briefly mentioned (e.g., PRA p. 21.5-9, DCD p. 19.3-20). Also, a corium splash shield is identified in PRA Figure 4.18-1, but is not mentioned or discussed anywhere in the DCD or PRA. Provide a more detailed discussion and evaluation of these features.
19.0.0-6	R. Palla	Provide expanded PRA uncertainty and importance analysis evaluation.	The assessment of PRA uncertainty and importance analysis (PRA Section 11) is superficial, and focuses largely on Level 1 results. Provide a substantially expanded evaluation, addressing the uncertainty and sensitivity of results to key containment-related features, assumptions, and operator actions (e.g., BiMAC availability and effectiveness, containment venting, use of drywell sprays, and isolation condenser makeup via AC-independent fire pumps).
19.0.0-7	R. Palla	Provide substantial information related to containment isolation provisions and failures.	The documentation of containment isolation provisions/failures (DCD Section 6.2 and PRA Section 4.18) is lacking a substantial amount of information needed to complete the staff review. For example, figures showing isolation valves and numbers for all containment penetrations have not been provided, numerous systems/penetrations identified in DCD Section 6.2 are not modeled in the PRA (e.g., isolation condenser purge line and excess flow line, standby liquid control system, containment inerting system chilled cooling water system), criteria for screening systems/penetrations from more detailed treatment in the PRA is not provided, the fault tree analysis considers only pipe breaks outside containment and does not include treatment of additional lines that might be open and need to be isolated, the system fault trees for steam suppression function failure (GT10-0001-_2) and reactor water cleanup (RWCU) isolation failure (GT10-0001-_7) are missing from Appendix B.4.18 of the PRA, and the RWCU valve numbers in Figure 4.18-1 of the PRA are inconsistent with those reported on Table 6.2-31 of the DCD. A review of this containment failure mode cannot commence until such additional information is provided. Please provide the requested information.

19.1.0-1	N. Saltos	Address passive system Thermal-Hydraulic (T-H) uncertainty.	Passive System Thermal-Hydraulic (T-H) Uncertainty Please address passive system T-H uncertainty. The issue of T-H uncertainty, also called passive system performance uncertainty, is not addressed in the ESBWR PRA. The issue of T-H uncertainty rises from the "passive" nature of the safety-related systems used for accident mitigation. Passive safety systems rely on natural forces, such as gravity, to perform their functions. Such driving forces are small compared to those of pumped systems and the uncertainty in their values, as predicted by a "best-estimate" T-H analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with frequency high enough to impact results, which are not predicted to lead to core damage by a "best-estimate" T-H analysis, may actually lead to core damage when T-H uncertainty is considered in the PRA models. T-H uncertainty, and its impact on PRA models, has been addressed in the certification of the AP600 and AP1000 designs through the use of a structured "margins" approach. This approach accounted for T-H uncertainty in the PRA by adopting conservative success criteria for safety systems and operator actions. It is stated in the submitted PRA for the ESBWR design that the issue of T-H uncertainty has been addressed "by increased design redundancies in the key passive systems and components." The staff believes that the issue of T-H uncertainty cannot be resolved by this statement alone, even though increased system redundancies may have a beneficial effect in addressing this issue. The accounting for T-H uncertainty may result in more conservative success criteria than those currently assumed in the PRA, which in turn will result in higher risk estimates. For example, the assumed success criteria for core cooling using the gravity driven cooling system could change to require the opening of two out of four equalizing lines instead of the one out of four lines currently assumed in the PRA. Such potential changes can have a significant impact on the results and insights of the PRA and could lead to additional design certification requirements, such as requirements for regulatory treatment of non-safety systems (RTNSS).

19.1.0-2	N. Saltos, M. Pohida	Provide documentation/analyses in support of the process used to identify requirements for RTNSS.	<u>Incomplete Documentation/Analyses in Support of the Process Used to Identify Requirements for RTNSS</u> The U.S. Nuclear Regulatory Commission and the Advanced Light-Water Reactor Steering Committee reached consensus on a process for resolving the RTNSS issue (SECY-94-084). This process included the use of both probabilistic and deterministic criteria to achieve the following objectives: (1) determine whether regulatory oversight for certain non-safety-related systems was needed, (2) identify risk important structures, systems and components (SSCs) for regulatory oversight (if it were determined that regulatory oversight was needed), and (3) decide on an appropriate level of regulatory oversight for the various identified SSCs commensurate with their risk importance. No adequate documentation of this process is provided in the submitted ESBWR PRA. Provide the following information: (a) Assessment of risk, in terms of both core damage frequency (CDF) and large release frequency (LRF), for external events at power and during shutdown assuming no credit for non-safety systems (focused PRA). This information is needed in the two probabilistic criteria (total CDF less than 1E-04/yr and total LRF less than 1E-06/yr). (b) A risk analysis supporting the RTNSS process at shutdown. As one would expect, failure of the non-safety related reactor water cleanup/shutdown cooling system (RWCU/SDC) would cause an initiating event and loss of the decay heat removal function. This condition may require additional regulatory treatment for the RWCU/SDC system and its non-safety related support systems because this system is not in the technical specifications and its failure drives the shutdown PRA results. It should be noted that Westinghouse had additional regulatory controls for the analogous non-safety residual heat removal system and its support systems in the AP1000 design in accordance with RTNSS criteria.

			(c) In applying the probabilistic criteria, the RTNSS process stresses the importance of accounting for uncertainties and also taking into consideration the risk importance of SSCs contributing to initiating event frequencies. No such information is provided in the submitted ESBWR PRA. (d) Results (dominant accident sequences and cutsets with associated frequencies) of the “focused” PRA sensitivity study must be submitted. Cutsets contributing to 90 percent of CDF and/or LRF or top 200 cutsets, whichever is smaller, are needed to provide adequate information for the staff’s review. Also, a discussion regarding the use of PRA results in the RTNSS decision-making process is needed (e.g., how it is decided whether regulatory oversight for certain non-safety systems is needed; how risk important SSCs for regulatory oversight are identified; and what is the basis for deciding on an appropriate level of regulatory oversight for these SSCs).
19.1.0-3	N. Saltos	Provide additional cutsets and a discussion on the use of uncertainty, sensitivity and importance analyses.	<u>PRA Results and Related Discussion</u> The documentation of CDF quantification results (in Section 7.0) provides only the top ten cutsets contributing to the internal events CDF for review. Provide cutsets contributing to 90 percent of CDF or top 200 cutsets, whichever is smaller, as the initial information for the staff’s review. In addition, provide a discussion on how the uncertainty, sensitivity and importance analyses are being used to provide insights and identify requirements for structures, systems, and components as well as for human actions.

19.1.0-4	N. Saltos, M. Pohida	Identify design certification requirements based on PRA insights and assumptions.	<u>Identify Design Certification Requirements Based on PRA Insights and Assumptions</u> The use of PRA results and insights to identify design certification requirements for the ESBWR design is an important objective of the certification process. These requirements aim at ensuring that PRA assumptions (e.g., regarding design features and operation of a safety system, system interactions and human actions) associated with risk important features will "come true" in a future plant referencing the ESBWR design and that uncertainties have been appropriately addressed. No such information is included in the submitted ESBWR PRA.
19.1.0-5	N. Saltos	Provide appropriate ESBWR data base references.	<u>ESBWR Data Base References</u> The component reliability data base (Tables 5-1 and 5-2) used in the submitted PRA makes extensive reference to the simplified boiling-water reactor (SBWR) and advanced boiling-water reactor PRA data bases. The SBWR design should not be referenced in ESBWR since it has not been reviewed and certified by the staff. Furthermore, the sources of information for a database should not be another PRA's database. Please provide appropriate data base references.
19.1.0-6	N. Saltos	Provide detailed evaluations of important human actions and their associated human error probabilities.	<u>Human Reliability Analysis</u> The scope of the human reliability analysis is limited to a preliminary analysis of human errors related to improper realignments following tests, maintenance, or calibrations (Type A), or human errors that affect various systems (Type C), and the use of generic values. The GE submittal indicated that detailed analysis of important human actions affecting the operation of the ESBWR design will be provided in the future. Detailed evaluations of important human actions and their associated human error probabilities must be provided for an expeditious review of the GE submittal.

19.1.0-7	N. Saltos	Address fire risk analysis issues.	Fire risk analysis The fire-induced vulnerability evaluation (FIVE) methodology and data for fire ignition frequency estimates were used in the fire risk analyses. The submitted ESBWR PRA does not provide adequate details on the screening out of non-risk significant fire areas/zones, nor the thresholds used in the screening process. In addition, no fire and smoke propagation into a second (adjacent) fire area is considered, even though there is a probability that penetrations in a fire barrier will fail and allow the fire to grow in adjacent areas. The assumption of no fire growth to an adjacent area, together with smoke propagation and the issue of fire-induced hot shorts of squib valves, need to be addressed in the fire PRA. Furthermore, the fire PRA should provide input, as necessary, to the RTNSS process. Addressing these issues may result in a significant revision of the submitted fire PRA.
19.3.0-1	M. Pohida	Provide risk assessment for fires and floods at shutdown.	Risk Assessment for Fires and Floods at Shutdown The fire and flood risk assessment for shutdown is missing from the PRA. Please provide a risk assessment for fires and floods at shutdown. The fire PRA used the Electric Power Research Institute FIVE methodology which only considers full power internal event risk. The very low ESBWR fire CDF was based on physical and electrical separation that may be breached at shutdown. The very low ESBWR flood CDF was based in part on the use of flood detection and the use of water tight doors. The water tight doors could be breached at shutdown and the flood detection equipment may not be available.
19.3.0-2	M. Pohida	Provide a discussion of large release frequency (LRF) risk at shutdown.	Large Release Frequency (LRF) Risk at Shutdown Even though release frequencies (core damage with the containment bypassed) appear to be quantified in Table 16.6-2, the shutdown PRA lacks a discussion on LRF risk. Please provide this discussion.