December 26, 2006

Mr. David H. Hinds, Manager, ESBWR
General Electric Company
P.O. Box 780, M/C J70
Wilmington, NC 28402-0780

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 88 RELATED TO 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 pursuant to 10 CFR Part 52. The 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. This RAI concerns NEDO-33201, Revision 1, ESBWR Probabilistic Risk Assessment.

RAIs: 19.1-66 through 19.1-116

To support the review schedule, you are requested to respond by January 31, 2007.

If you have questions or comments concerning this matter, please contact me at (301) 415-0224 or tak@nrc.gov or you may contact Amy Cubbage at (301) 415-2875 or aec@nrc.gov.

Sincerely,

/RA/

Thomas A. Kevern, Senior Project Manager
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure: As stated

cc: See next page

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ACCESSION NO. ML063400331

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DATE	12/22/2006	12/26/2006

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Requests for Additional Information (RAIs)
NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment (PRA)"

RAI number	Reviewer	Summary	Full Text
19.1-66	Saltos N	Clarify/explain modeling of the Containment System (Section 4.18).	The staff needs additional information, related to Section 4.18 (Containment System), in the following areas: (A) It is stated (page 4.18-3): "The Containment Isolation System provides protection against release of radioactive materials to the environment as a result of accidents occurring in systems or components within the containment." However, the next paragraph seems to contradict this statement. It is stated that "In the analysis, only the isolation of LOCAs outside containment....... .. are justified." Please clarify. (B) It is stated (page 4.18-3): "Each vacuum breaker is equipped with a DC motor-operated valve which provides isolation capability if the vacuum breaker sticks open or leaks in the closed position." This statement does not seem to be in agreement with the fault tree modeling of the "failure to isolate vacuum breaker (VB) leaks" top event (GT10-0001_1). Instead of modeling the isolation failure of the DC motor-operated isolation valves, the failure to close of the VBs themselves is modeled. Please explain. (C) Fault trees for two top events related to VBs are included in Appendix B.4.18-1. Top event GT10-0001-_1 models the failure to isolate VB leaks (needed to model failure of passive containment cooling, which requires leak-tight closure of all three VBs to operate efficiently). Top event GT10-0001-_2 models the failure of the steam suppression function, which requires that at least two VBs remain closed (for efficient steam suppression) and at least one VB opens when demanded to relieve pressure. However, it appears that no VB leaks during steam suppression were considered. If there is a leak in one VB and it is isolated, this VB will not be available to open to relieve pressure (which may be needed more than once during the steam suppression phase). Please explain why potential VB leakages during steam suppression were not considered. (D) Section 4.18-7 (Basic Events) appears to have several errors. For example, the description provided for event B32-ACV-OO-F001A is "MSIV F001A Valve Fails to Close" instead of "Isolation Condenser Steam Line F001A Valve Fails to Close." Another example is the definition of event B32-MOV-OO-F002A as "Solid State Load Driver LD002 fails to Close"

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			instead of "Isolation Condenser Steam Line MOV F002A Fails to Close." Please review the basic event descriptions provided in Section 4.18-7 and revise as necessary. (E) In Section 4.18-2 (System Dependencies Matrix), no information is provided for several components (e.g., air-operated valve G31-F3B and the eight Main Steam Isolation Valves (MSIVs)). Please review the information provided in Section 4.18-2 and revise as necessary. (F) It is stated that there is a third normally open valve (G31-F3A or B) in the suction line of each train of the reactor water cleanup (RWCU) system which is air-operated. These valves are not shown in the simplified flow diagram but are credited in the PRA. Please provide information about the location of these valves and explain why they are not part of the system boundary, as indicated in Section 4.18.9.2 (page 4.18-6). Are these valves supplied by individual nitrogen or air accumulators? (G) It is stated that top event GT10-0001-_9 represents the failure to isolate main steam pipe breaks outside containment. However, as modeled, this top event represents the failure to isolate all four main steam lines and not just the line where the break is located. Please explain and state any important assumptions that were made in the event tree for main steam line breaks outside containment. Also, clarify the group of MSIVs included in basic event B21-ACV-CF-MSIVCLOSE (e.g., all eight valves versus two valves in same line) and state any assumptions made in modeling common cause failure of MSIVs. (H) For other systems modeled in the PRA, provide a simplified instrumentation and control (I&C) block diagram, with a brief description of each block element, that supports the logic of the fault trees included in Appendix B.4.18.
19.1-67	Saltos N	Clarify/explain modeling of the Passive Containment Cooling System (Section 4.19).	The staff needs additional information related to Section 4.19, Passive Containment Cooling System (PCCS), in the following areas: (A) it is stated (Section 4.19.8) that "The pool water in each . . . subcompartment is removable without emptying the entire ICS/PCCS pool. The individual partitioned PCC (passive containment cooling) pool is isolated by closing the locked open valve . . . to replenish the pool, the normally open valve is re-opened and the water refills the pool." If this task is allowed to be performed during plant operation at power, PCC pool unavailability due to

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			maintenance should be included in the PRA models as well as operator failure to open the valves. Similarly, if online corrective maintenance of PCC condensers is allowed by the technical specifications, the unavailability of the impacted IC (isolation condenser) loop(s) should be modeled. Please explain. (B) It appears that a potential failure, which could also impact multiple PCC loops (common cause failure), is the plugging of the spargers in the suppression pool. Was such a potential failure investigated? Please discuss. (C) The fault tree provided in PRA Appendix B.4.19 (page 1) for "loss of pool water during 72 hours" includes two identical descriptions for events GB32-0201-_1 and GB32-0301-_1. Please clarify by providing descriptions consistent with Figure 4.19-2 (Schematic of IC/PCC Pools and Interconnections). Also, it appears that not all common cause failure combinations of motor operated valved (MOVs) F72H A,B,C,D (consistent with PCCS system success criteria) were considered. Please explain. (D) The PCCS drain lines discharge into the Gravity Driven Cooling System (GDCS) pools. Located on the drain line and submerged in the pool, just upstream of the discharge point, is a loop seal which prevents back-flow of steam and gas mixture from the drywell (DW) to the vent line (such back flow could short circuit the flow through the condenser to the vent line). The potential that the function of this feature (loop seal) is defeated should be investigated. Does this feature prevent back-flow under all accident conditions and in the presence of multiple failures? For example, do all seal loops prevent back-flow during a large LOCA with failure of multiple GDCS lines to inject? Please discuss.
19.1-68	Saltos N	Discuss the dominant accident sequences and associated major contributors and interpretation of results (Section 7).	Section 7 of the PRA documents the results of core damage frequency (CDF) due to internal initiating events that occur when the plant is operating at power. This documentation is mostly in tabular form without much discussion and interpretation of the results. A discussion of the dominant accident sequences and associated major contributors to CDF, as well as an interpretation of the results in terms of important design and operational features, should be included. Such information is one of the inputs to the integrated process used to gain insights about the design and identify design and operational requirements. The discussion should include:

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			(1) a characterization of the dominant accident sequences and associated major contributors to the sequence CDF; (2) a characterization of the major contributors to the uncertainty associated with the estimated CDF; and (3) a characterization of the major design and/or operational features that contribute to the reduced CDF risk of the proposed design as compared to operating designs (e.g., less reliance on offsite and onsite power, and divisional separation). Since the ESBWR design has evolved from current BWR technology, through the incorporation of several features intended to make the plant safer, more available and easier to operate, the results of the risk evaluation should indicate that the design represents a reduction in risk over existing plants. For this purpose, a generic qualitative comparison of risks, by initiating event category, between the proposed design and operating BWR plants helps identify major design features that contribute to the reduced risk of the proposed design as compared to operating designs. Please provide this information.
19.1-69	Saltos N	Expand PRA results in Section 7 to include post 24-hour system or operator actions needed to prevent core damage.	In general, an accident sequence is categorized as successful if the reactor achieves a stable shutdown condition without core damage and this condition can be maintained for at least 24 hours following event initiation without further operator action or system operation. This means that it is not sufficient to avoid core damage during the first 24 hours if conditions are not stabilized in 24 hours, or if core damage is anticipated following 24 hours without any action. The results provided in Section 7, which are used to perform sensitivity and importance analyses (documented in Section 11), do not include post 24-hour system or operator actions needed to prevent core damage. For example, the successful post 24-hour operation of ICS and PCCS require the opening of a pair of motor-operated valves MOVs to replenish the IC/PCC pools. The staff could not find any cutsets including common cause failure of these MOVs. The modeling of post 24-hour actions in the PRA can be very important in sensitivity studies, such as the “focused PRA” sensitivity study which is used to identify non-safety-related systems that are candidates for regulatory oversight. Please discuss.

RAI number	Reviewer	Summary	Full Text
19.1-70	Saltos N	Address sensitivity analysis regarding 72-hour mission time and core cooling vulnerability event trees (Section 11).	The staff needs additional information, related to Section 11.3.1 (Mission Time), as follows: (A) It is stated (page 11.3-2) that for accident sequences, resulting from the core cooling vulnerability (CCV) event trees and which are grouped into the "ok" category, core damage can be avoided for much longer than the 72-hour mission time. If the means used in the CCV event trees are intended to prolong core cooling until other means become available or actions are taken, then such means and/or actions need to be characterized and discussed. (B) It is stated (page 11.3-2) that the CCV event trees for transients include mainly the Fuel and Auxiliary Pool Cooling System (FAPCS) and the Fire Protection System (FPS) as additional active cooling systems to be used in sequences involving class II accidents after successful water injection from GDCS and the suppression pool (top event VLFL). However, these sequences already include failure of the FAPCS in the suppression pool cooling mode. Therefore, it is likely that the FAPCS will not be available for injection. Please explain. (C) It is stated (page 11.3-2) that in the CCV event tree for loss of preferred power (LOPP) credit is taken for recovery of offsite power within 24 hours assuming failure to recover power at 30 minutes. A conditional failure probability to recover offsite power within 24 hours of about 10 percent is assumed. Please justify this assumption.
19.1-71	Saltos N	Address sensitivity analysis associated with non-safety-related systems (Section 11).	The staff needs additional information, related to Section 11.3.2 (Importance of Non-Safety Systems), in the following areas: (A) It appears that the cutsets provided in Section 11-5 are based on a mission time of 24 hours and important failures occurring after 24 hours from the initiation of the accident are not considered. For example, the successful post 24-hour operation of ICS and PCCS require the opening of a pair of MOVs to replenish the IC/PCC pools. The staff could not find any cut sets including common cause failure of these MOVs. If such failures are considered, the calculated CDF and LRF values in the "focused PRA" sensitivity study, which are used to identify non-safety-related systems that are candidates for regulatory oversight, could be much higher than those reported in the PRA. Please explain.

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			(B) The systems that are considered unavailable in the "focused PRA" sensitivity study are listed on page 11.3-3. This list does not include any support systems, such as onsite power, cooling water and non-safety-related I&C. Please explain. (C) The results of the "focused PRA" sensitivity study include significant uncertainties (related to both modeling assumptions and data) that need to be addressed and taken into account in identifying candidate systems for regulatory oversight. Please discuss.
19.1-72	Saltos N	Address results and insights from the uncertainty, importance and sensitivity analyses (Section 18).	Section 11 includes the results of sensitivity analyses. Additional sensitivity analyses may be needed to address open issues identified by the staff's review or due to changes in the design or the PRA. Section 18 includes the results of the importance analyses in tabular form without much discussion, interpretation or explanation of how these results have been used in an integrated fashion to support the design certification process. A discussion is needed to explain how the results and insights from the uncertainty, importance and sensitivity analyses have been used systematically, in an integrated fashion, to gain insights about the design and identify appropriate requirements to ensure that important assumptions made in the risk evaluation will remain valid in a future plant referencing the certified design and that uncertainties have been appropriately addressed. Please discuss.
19.1-73	Saltos N	Address uses of PRA in the design process (Section 18).	Please include a discussion on the use of PRA in the design process, with representative examples, of ways in which the ESBWR design was enhanced by adding or modifying design features or operational requirements. Please provide at least one example in each of the following three categories, if available. • Use of PRA to identify and introduce features and requirements in the ESBWR design that reduce or eliminate known vulnerabilities (or weaknesses) in operating BWR designs. • Use of PRA to quantify the effect of new design features and operational strategies on plant risk to confirm the risk reduction credit for such improvements. • Use of PRA to select among alternative features, operational strategies, or design options.

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19.1-74	Saltos N	Provide plant layout drawings for fire areas and fire boundaries and identify equipment (Section 12).	Please provide (1) plant layout drawings showing fire area boundaries and equipment located in each fire area, and (2) list of equipment located in each fire area (including cables routed through the fire area). This information is needed to clarify certain statements and assumptions made in the fire risk analysis (Section 12 of the PRA). For example, it is stated that the pumps and the heat exchangers of each RWCU train are located in different fire areas without stating in what fire areas they are located and through which fire areas the cabling to the RWCU pumps are routed. Also, please provide a list of any fire areas that were screened out from detailed analysis and discuss the basis. For example, it appears that fires in the yard area causing loss of offsite power (frequency 1.5E-2/year) and the remote shutdown panels were not addressed. Please clarify.
19.1-75	Saltos N	Address propagation of smoke to areas beyond the postulated fire (Section 12).	Smoke damage of advanced digital I&C system components can prevent actuation of multiple components. This issue has not been addressed in the fire PRA (Section 12). The propagation of smoke (e.g., through the ventilation system) to certain areas of the plant could impact equipment divisions supporting redundant functions and result in significant risk increases. Please address this issue by: (1) searching for potential smoke propagation paths; (2) identifying design and operational features that are available to prevent or minimize smoke propagation; and (3) assessing the associated risk or showing (e.g., through bounding assumptions) that such risk is insignificant. Smoke propagation paths between safety-related areas as well as between a safety-related and a non-safety related area should be investigated. Please discuss.
19.1-76	Saltos N	Address risk importance of non-safety-related systems in the fire PRA (Section 12).	Please provide the risk importance measures for non-safety-related systems that were credited in the fire risk assessment. The conservative assumptions used in the fire risk analysis do not provide insights regarding the importance of non-safety-related systems to mitigate accident sequences initiated by fire events.

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19.1-77	Saltos N	Investigate fire-induced spurious valve actuations causing LOCA or incorrect valve lineup (Section 12).	The staff believes that a systematic search of potential fire-induced spurious valve actuations (single and multiple) causing LOCA or incorrect valve lineup is needed and should be addressed in Section 12 of the PRA. Fires have the potential to cause spurious valve actuations. Smoke damage of advanced digital I&C components can prevent actuation of multiple components. Although smoke-induced scenarios are not routinely treated in conventional PRAs, they should be investigated in this case because of the potential vulnerability of advanced electronics to smoke damage. If such failures are found to be believable, based on frequency estimates, their risk impact should be assessed. Also, the spurious opening of containment isolation valves as well as the opening of valves in paths between the reactor coolant system and low pressure systems (leading to interfacing system LOCAs) should be investigated. Please discuss any ESBWR features that prevent or mitigate spurious actuation and smoke damage (and their impacts, as necessary) of I&C components.
19.1-78	Saltos N	Address probability of fire barrier failure and propagation of fire to an adjacent area (Section 12).	It is stated that "A value of 7.4E-03 is taken as the fire propagation probability from one divisional fire area to another. This probability represents failure of a fire door and is obtained from Reference 12-4." However, there may be more than one fire door separating the two fire areas and additional fire propagation pathways, such as piping or cable penetrations and ventilation ducts (especially in large "fire areas" assumed in order to simplify the analysis). Reference 12-4 (listed in the submitted fire risk analysis) provides fire barrier failure probabilities associated with such fire propagation pathways. In addition, fire doors in some fire areas can be open to perform online maintenance. Please provide additional justification of the assumed probability of fire barrier failure by addressing each of these staff concerns.
19.1-79	Saltos N	Address assumptions regarding main control room fires (Section 12).	Verify, explain and/or clarify the following assumptions regarding main control room (MCR) fires: (A) Verify that if MCR evacuation is necessary, the remote shutdown panels provide complete redundancy in terms of control and monitoring for safe shutdown functions. (B) Explain how the transfer of operations from the MCR to the remote shutdown panels is controlled (e.g., is there a transfer switch and associated analog-to-digital converter?). Explain why a fire-induced short or flash lightning cannot generate spurious actuation signals. Please list the features of the proposed optical fiber design that prevent spurious actuations, given a fire in the MCR or the remote shutdown panels.

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			(C) Verify that the MCR and remote shutdown panel are in separate fire areas and each has its own dedicated ventilation system. (D) List any design and operational features that limit MCR fire ignition frequency (e.g., low voltage, low current equipment, fiberoptic cables, and administrative procedures). (E) List any design and operational features that prevent or limit the propagation of smoke from MCR fires to adjacent areas (including the remote shutdown panels). (F) Discuss the potential of spurious actuations associated with hard-wired controls.
19.1-80	Saltos N	Address opening of Safety Relief Valves from fire in Reactor Building.	Please explain why a fire in the Reactor Building Divisional Zones is not likely to cause the inadvertent opening of multiple SRVs. Describe the actuation mechanism of SRVs, their power sources and the separation of power and control cables associated with the various SRVs (Section 12).
19.1-81	Pohida M	Assess risk of open containment in Mode 5 in the Shutdown PRA.	In Section 16.2.1.2, Cold Shutdown, the PRA states that the containment is opened at some time during Mode 5, but since it is intact most of the time, the PRA assumes containment to be intact. If containment can be opened in Mode 5, the PRA should be revised to assume an open containment. Please clarify and/or revise the PRA, as needed.
19.1-82	Pohida M	Assess risk of de-inerted containment for Mode 4 in Shutdown PRA.	In stable shutdown, Mode 4, containment can be de-inerted, but containment integrity is required by Technical Specifications. The CDF contribution in this mode is assumed to be included in the Full Power PRA. The staff understands from DCD Chapter 6.2.1.1, that the containment is inerted during power operation to prevent the formation of a combustible mixture if oxygen were present following a severe accident. If containment integrity cannot be assured following a severe accident in Mode 4, then mode 4 should be modeled separately. Please clarify and/or revise the PRA, as needed.
19.1-83	Pohida M	Assess de-inerted containment in Modes 5 and 6 when reactor vessel head is on.	The PRA should address the risk of a failed containment during Modes 5 and 6 when the reactor vessel head is on. Please clarify and/or revise the PRA, as needed.

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19.1-84	Pohida M	Address Mode 5 definitions in the PRA and Technical Specifications.	In section 16.2.1.2, the PRA states, “Initial RPV conditions in this mode are a pressure of .75 MPA (109 psia) and a temperature of 334°F.” This statement is inconsistent with the definition of cold shutdown as defined in Technical Specifications Table 1.1-1, dated 2/28/06. Please clarify and/or revise the PRA, as needed.
19.1-85	Pohida M	Assess LOCAs during Mode 5 with open containment.	Section 16.3.1.2.1.1 of the PRA indicates that no specific shutdown LOCAs are required for shutdown modes where the reactor vessel is closed. LOCAs during Mode 5 should not be included in the full power PRA and should be assessed separately considering the containment can be open and/or de-inerted. Please clarify and/or revise the PRA, as needed.
19.1-86	Pohida M	Assess valve mis-alignments and operator-induced coolant losses in Shutdown PRA.	The staff reviewed P&IDs for the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC), evaluated piping penetrations in the reactor vessel bottom head upstream of containment isolation valves in the RWCU/SDC, and found numerous piping connections to low conductivity waste and the process sampling system. To justify that draindown events do not need to be quantified in the shutdown PRA, please document: (1) the sizes of the lines in the reactor vessel bottom head and (2) the administrative controls necessary to prevent these lines from becoming RCS draindown paths from operator error.
19.1-87	Pohida M	Address initiating event frequencies for reactor pressure vessel leaks and diversions.	The initiating event frequency used in the Shutdown PRA is based only on pipe breaks. However, EPRI derived an initiating event frequency for RPV leaks or diversion of 2.8E-5 per hour (Table 7-3 of "An Analysis of Loss of DHR Trends and IEF (1989--2000), EPRI 1003113, November 2001) and the data for this initiating event frequency does not include pipe breaks. Please assess and document in the shutdown PRA the additional initiating event frequency contribution from non-pipe breaks to reflect actual industry experience.

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19.1-88	Pohida M	Assess drain down paths outside containment for Modes 5 and 6.	The PRA states that breaks outside containment can originate only in RWCU/SDC piping, as this is the only system that moves reactor coolant from the containment in Mode 6; the rest of the RPV piping is isolated. The RWCU/SDC containment penetrations have redundant and automatic power-operating containment isolation valves that close on signals from the leak detection and isolation system and the reactor protection system. However, the Technical Specifications (dated 2/28/06) identify that isolation instrumentation discussed in Section 3.3.6.1 is only required in Modes 1-4 and, as stated in Section 3.6.1.3, the containment isolation valves are only required to be operable in Modes 1-4. Since containment isolation valves are not required to be operable in Modes 5 and 6, the Shutdown PRA does not include the risk of reactor vessel drain downs through the RWCU/SDC outside containment. Please clarify and/or revise the PRA, as needed.
19.1-89	Pohida, M	Address reactor vessel draining during replacement of fine motion control rod drives.	As stated in the PRA, the potential exists for an operator to remove the control rod blade inadvertently, establishing a direct path for draining the RPV during fine motion control rod drive (FMCRD) maintenance. The PRA should document the controls (e.g. Technical Specifications) that prevent the operator from: (1) failing to install the blind flange and (2) failing to recognize that the blade to be pulled out is withdrawn and already decoupled.
19.1-90	Pohida M	Address operator closure of both equipment hatches following a LOCA at shutdown.	The PRA credits the operator in closing both containment hatches before RCS overflows through the hatches (water level is below the bottom edge of the hatch). Please document in the PRA whether equipment hatch closure takes place inside primary containment or outside primary containment. This detail is important in understanding how the human error probabilities can be quantified. Current models for operator actions may not be applicable if the operator has to close the equipment hatch from inside primary containment while the lower drywell is flooding.
19.1-91	Pohida M	Address isolation condenser actuation during plant shutdown.	In the PRA, the Isolation Condenser System (ICS) function is credited in the events for losses of decay heat removal in Mode 5. According to Technical Specifications (dated 2/28/06), the ICS is required to be operable, but the staff could not find the operability of the ICS instrumentation and/or the operability of the ICS in Sections 3.3 and 3.4. Please clarify and/or modify the PRA to reflect that operator action is needed to initiate the ICS.
19.1-92	Pohida M	Address Control Rod Drive injection during Modes 5 and 6.	In section 16.4.1.1. of the PRA, automatic high pressure injection using Control Rod Drive (CRD) pumps is credited on RPV water level 2. However, automatic actuation of CRD on level 2 is not covered in Technical Specifications (dated 2/28/06) for Modes 5 and 6. Please clarify and/or revise the PRA to reflect that operator action is needed to initiate the CRD pumps.

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19.1-93	Pohida M	Address operability of Safety Relief Valves for Modes 5 and 6 when reactor vessel head is on.	In section 16.4 of the PRA, alternate decay heat removal using the Safety Relief Valves (SRVs) is credited in many shutdown success paths. However, operability of the SRVs is not covered in Technical Specifications (dated 2/28/06) in Modes 5 and 6 when the reactor vessel head is on. Please clarify and/or revise the PRA to reflect that the SRVs may not be available for decay heat removal.
19.1-94	Pohida M	Address opening of Safety Relief Valves and/or Depressurization Valves for shutdown decay heat removal.	In section 16.4 of the PRA, the opening of the SRVs is credited for enabling low pressure makeup using the Fuel and Auxiliary Pool Cooling System (FAPCS) or the Fire Protection System (FPS). Additionally, the opening of the Depressurization Valves (DPVs) is credited for enabling GDCS. Please revise the PRA to address whether operator action is needed to open the SRVs and/or DPVs.
19.1-95	Pohida M	Address use of the Automatic Depressurization System for short term and long term cooling.	In Section 16.4, the PRA credits automatic short term and long term cooling using 2 out of 8 lines of the Gravity Driven Cooling System (GDCS), 2 out of 3 GDCS pools and the opening of at least one equalizing line. However, depressurization is accomplished by the Automatic Depressurization System (ADS) and operability of ADS is not covered in Technical Specifications (dated 2/28/06). Please clarify and/or revise the PRA to reflect that operator action is needed to initiate ADS, and the ADS function may not be operable.
19.1-96	Pohida, M	Address thermal-hydraulic uncertainty for short term and long term core cooling.	To address thermal-hydraulic uncertainty regarding shutdown success criteria, please provide additional information (e.g., summary and results of calculations) that justifies short term and long term core cooling using (1) 2 SRVs, (2), 2 out of 8 lines of GDCS, (3) 2 out of 3 GDCS pools, (4) the opening of at least one equalizing line, and (5) the opening of 4 depressurization valves (DPVs) during Mode 5 when the reactor vessel head is on.
19.1-97	Pohida M	Explain restart of Reactor Water Cleanup/Shutdown Cooling System pumps.	Please clarify in the PRA whether the RWCU/SDC restarts automatically after a loss of preferred power (LOPP) event following start up of the non-safety related diesel generators.
19.1-98	Pohida M	Address human error probability in shutdown PRA.	Numerous operator actions are discussed in the PRA, but the PRA does not address how failure of these operator actions were quantified. Please clarify and/or revise the PRA to address how the human error probabilities were estimated.

RAI number	Reviewer	Summary	Full Text
19.1-99	Pohida M	Provide a human error sensitivity analysis for the Shutdown PRA.	The staff needs additional information to understand the role of the operator in preventing and mitigating shutdown events to prevent core damage. Please provide a human error sensitivity analysis for the Shutdown PRA. For example, assume all operator actions failed (i.e., human error probability is 1.0) and assume operator actions have a failure rate of .01.
19.1-100	Pohida M	Address common-cause failure of non-safety related components for loss of shutdown cooling.	As described in PRA Section 20.4.4.6, Loss of Shutdown Cooling was excluded from the initiating events assessment since both trains of RWCU/SDCS need to fail to cause a loss of the decay heat removal function. Common cause failure of the non-safety related RWCU/SDC pumps or the common cause failure of the non-safety related Reactor Component Cooling Water System (RCCWS) pumps was not considered. Please revise the RTNSS evaluation to consider common cause failure of non-safety related components associated with RWCU/SDCS and its support systems for the shutdown initiating events evaluation.
19.1-101	Pohida M	Address isolation of containment penetrations for Modes 4 - 6.	As described in PRA Section 20.4.4.7, Shutdown LOCA was excluded from regulatory oversight based on automatic isolation of RWCU/SDCS and FAPCS containment penetrations which are not required by Technical Specifications. Please revise the RTNSS evaluation to reflect that the Technical Specifications do not require these isolations to be operable during Modes 4, 5, and 6.
19.1-102	Saltos N	Provide information for flooding areas considered in the flooding risk analysis (Section13).	Please provide (1) plant layout drawings showing flooding area boundaries, elevations and adjacent areas, as modeled in the flooding risk analysis, and (2) list of equipment credited in the PRA for accident mitigation or contributing to accident initiation that is located in each flooding area (specify whether the area includes safety-related equipment or non-safety-related equipment and include assumptions regarding cables routed through each flooding area below the maximum expected flood height). This information is needed to understand or clarify statements and assumptions made in the flooding risk analysis (Section 13).
19.1-103	Saltos N	Address areas that were screened out from the detailed flood risk analysis (Section13).	Please provide a list of areas that were screened out from detailed flooding analysis and discuss the basis. Also, please identify any flooding sources which are located in analyzed areas but they have not been considered in the risk analysis (e.g., potential breaks in a GDCS injection line, an equalizing line, or a deluge line within the containment) and provide the basis.

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19.1-104	Saltos N	Explain the basis and assumptions for frequencies in flooding risk analysis (Section13).	In Section 13, the assumed flooding frequencies (for each flooding area) appear to be based on the frequency of a "limiting" source and not all sources of water. For example, it is stated (page 13.2-1) that *"The frequency of flood scenarios ...are based on generic information......The systems inside each building that could represent a flood source are considered. From these systems, the building flood source that presents the most critical characteristics for flood progression and which has the capacity to damage mitigation equipment is chosen."* Also, in Table 13-1 it is stated that the frequency of flooding in the Turning Building is based on a Cooling Water System (CWS) break. The staff believes that the flooding frequency should be based on all potential sources in each area and not just the source that causes the most damage (such assumption is adequate for the bounding treatment of the consequences of flooding). Please explain and clarify, as necessary.
19.1-105	Saltos N	Address flooding-induced failures of environmentally-qualified equipment (Section13).	The assumptions on page 13.2-1 include several (# 13, 15, 16, and 18) related to environmental qualification of electrical components, such as cables, connections, terminations, and junction boxes. No failures (with limited exceptions) of environmentally qualified electrical components, due to spraying or immersion, is considered in the flooding risk analysis. Please clarify and provide the basis for assuming that flooding-induced failures of environmentally qualified electrical components are negligible.
19.1-106	Saltos N	Clarify flooding from main steam and feedwater pipes located in the steam tunnel (Section13).	On page 13.5-3 is stated that *"The main steam and feedwater pipes are located in the steam tunnel. The water released by these breaks propagates toward the Turbine Building without affecting components located inside the Reactor Building. Therefore these flooding scenarios are addressed in the Turbine Building analysis..."* Please explain the design features (e.g., watertight steam tunnel capable to withstand the maximum anticipated hydrodynamic loads) which ensure that water from a break in main steam and feedwater pipes will be directed to the Turbine Building.
19.1-107	Saltos N	Clarify flood height, flood propagation and location of equipment (Section13).	For each flooding area considered in the flooding risk analysis (Section 13), please discuss the maximum expected flood height, flood propagation potential (e.g., through penetrations, open doors or under doors and down stairwells), and the location of equipment with respect to the maximum expected flood height.

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19.1-108	Saltos N	Address flooding risk analysis and internal events analysis related to support systems (Section13).	Please explain the basis of the assumption (page 13.5-7) that floods caused by breaks in several of the support systems (e.g., Plant Service Water System) have the same consequences as failure of the systems themselves and, since the consequences were already considered in the internal events analysis, such events are not further analyzed in the flooding analysis.
19.1-109	Saltos N	Address modeling of post 24-hour failures and actions in flooding risk analysis (Section13).	The PRA (Section 13) does not include modeling of post 24-hour failures and actions in the flooding risk analysis. The staff believes such modeling is important in sensitivity studies, such as the “focused PRA” used to identify non-safety-related systems that are candidates for regulatory oversight. Please include post 24-hour modeling or discuss the rationale for excluding.
19.1-110	Saltos N	Address risk importance of non-safety-related systems in flooding PRA (Section 13).	For the flooding PRA (Section13, operation at power), please provide the risk importance measures for non-safety-related systems that were credited in the flooding risk assessment. The conservative assumptions used in the flooding risk analysis do not provide insights regarding the importance of non-safety-related systems to mitigate accident sequences initiated by flooding events.
19.1-111	Saltos N	Discuss isolation of break in circulating water system and failure probability (Section13).	Please clarify how a break in the circulating water system (CIRC) is isolated. (e.g., number of automatically actuated valves that can isolate the break). Please explain why the probability for the failure to isolate the break used in the quantification (event tree of Figure 13-2) is an order of magnitude smaller than the probability of the same event reported in Table 13-4.
19.1-112	Saltos N	Provide additional insights gained from flooding risk analysis (Section13).	Please provide more detailed insights gained from the flooding risk analysis by linking the results and assumptions of the analysis to specific features of the design and planned operation. Insights from the flooding analysis which are reported in Section 13.7 are high level. The staff needs detailed insights about specific design and operational features to support the design certification.

RAI number	Reviewer	Summary	Full Text
19.1-113	Saltos N	Address assumptions in seismic margins risk analysis (Section15).	Please discuss the assumptions related to the seismic event and fault trees. The assumptions should provide the basis for ensuring that all important seismic and mixed (seismic and random) scenarios are addressed. Explain why no seismically-induced LOCAs (various sizes and locations) or loss of preferred power transients were considered in the analysis. Explain why no seismic failure of I&C components are considered. It appears that important assumptions about the seismic capacity of several structures, systems and components (SSCs) are not listed in Table 15-11 and are not shown explicitly in the seismic fault trees. For example, it is stated (page 15.4-3) that *"Failure of the standby liquid control system (SLCS) is dominated by failure of two components: squib valves and boron supply tanks."* This implies that other SLCS SSCs are assumed to have higher seismic capacities. In addition, please include the High Confidence of Low Probability of Failure (HCLPF) values in Table 15.4-3 for SSCs for which assumptions are made in the seismic analysis.
19.1-114	Saltos N	Clarify human actions and random failures in seismic margins risk analysis (Section15).	Please clarify the discussion (page 15.4-2) regarding screening out human actions and random failures. It is stated: *"Human actions are required only in the long term and......do not dominate system failure. As such, random failures are assumed to be non-significant ... ."* Please discuss whether any human actions are credited in the long term to recover from seismic failures and explain the correlation with the statement that random failures are not significant.
19.1-115	Saltos N	Clarify crediting of passive safety systems in the seismic margins risk analysis (Section15).	The statement (page 15.4-2) that only passive safety systems are credited in the seismic event tree of the seismic margins risk analysis is confusing. The seismic event tree for operation at power (Figure 15.2) includes the Fire Protection System (FPS) pump, which is not passive but has its own dedicated diesel generator. Please clarify and list HCLPF values for pump supporting equipment and piping.
19.1-116	Saltos N	Address design and operational requirements derived from PRA risk insights and assumptions (DCD Chapter 19).	In Table 19.2-3 (Risk Insights and Assumptions) of Design Control Document (DCD) Tier 2, Chapter 19, please include a list of the design and operational requirements (e.g., ITAACs, Technical Specifications, reliability assurance program, COL action items) that can be identified through a systematic search of risk insights and assumptions in the PRA. Identification of such requirements is one objective pertaining to use of the PRA. The staff recognizes that identification of such requirements is an iterative process and the list cannot be finalized until all analyses, and staff review/evaluation, are completed and open issues are resolved. Please discuss.

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