September 13, 2006

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 59 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 the ESBWR Probabilistic Risk Assessment and Chapter 19 of the ESBWR Design Control Document. These questions were sent to you via electronic mail on July 13 (19.5-3 thru 13), 23 (19.5-14), and 30 (19.1-20 thru 41), 2006, and were discussed with your staff during telecons on August 17 and 23, 2006. You agreed to respond to this RAI on the following schedule:

October 6, 2006: 19.1-22, 24, 25, 27 thru 30, 32 thru 34, 36, 38, 40, and 41.
October 27, 2006: 19.5-3 thru 14.
November 28, 2006: 19.1-20, 21, 23, 26, 35, and 39.
December 14, 2006: 19.1-31 and 37.

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
Division of New Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 52-010

Enclosure: As stated

cc: See next page

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cc: See next page
ACCESSION NO. ML062500194

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DATE	09/08/2006	09/13/2006

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Requests for Additional Information (RAIs)
ESBWR Design Control Document (DCD) Chapter 19
NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment," Sections 3, 4 and 5

RAI Number	Reviewer	Question Summary	Full Text
19.1-20	Saltos N	Provide definitions, tables and simplified diagrams in Section 4.5.	Please define acronyms, such as SSLC, RTIF and DPS, when they are first used in the text in Section 4.5 of NEDO-33201, and avoid using designators, such as B32, instead of the system name. Section 4.5 should be clarified by providing definitions, tables and simplified diagrams, to facilitate staff review of how the instrumentation and control (I&C) system was modeled in the PRA.
19.1-21	Saltos N	Provide simplified block diagrams for the Instrumentation, Logic and Control System, as modeled in the PRA.	Please provide simplified block diagrams of the Instrumentation, Logic and Control System supporting the fault tree analysis reported in Section 4.5.9 of NEDO-33201. Such block diagrams should show the processing of signals for both automatic and manual actuation of components and for the reactor trip function. Please include a description of each element and important assumptions made in the PRA model. This information is needed for the staff to understand how the I&C systems were modeled in the PRA.

RAI Number	Reviewer	Question Summary	Full Text
19.1-22	Saltos N	Provide basis for failure data used in I&C fault trees (Section 4.5) and discuss modeling of common cause failure of digital I&C software.	NEDO-33201, Tables 4.5-5, Common Cause Failure (CCF), and 4.5-7, List of System Basic Events, are included in Section 4.5 without any discussion. It is not clear to the staff how some of the probabilities reported in these two tables were obtained. Examples are: (1) the basis for the probability of basic event C51-ACT-LO-CHASRNM (1.90E-02); (2) the reason some basic events with same code (e.g., VLU-FC) have different failure probabilities; and (3) the reason that some CCFs, involving software failure, mentioned in Section 5.3.5 (e.g., C74-DTM-CF-RTIFALL) are not included in Table 4.5-5. (A) Provide complete lists of I&C basic and CCF events and discuss how the associated failure probabilities were calculated. (B) Discuss how the potential for CCF (both hardware and software) was investigated. Include discussion on the following: - The potential for software failure within subsystems and among cards; - Sensor and transmitter miscalibration; - Loss of cooling ventilation; - Manufacturing and installation errors; - Earthquake and fire; - Setpoint drift or incorrect setpoint; - Maintenance and test errors; and - Electromagnetic interference.
19.1-23	Saltos N	Explain or clarify several assumptions made in identifying and categorizing initiating events.	Explain or clarify assumptions in identifying and categorizing initiating events as follows: (A) The spurious actuation of all safety relief valves (SRVs) and all depressurization valved (DPVs) are considered and classified as large loss of coolant accidents (LOCAs). However, the spurious actuation of more than one but less than all is not considered. How was the frequency of such spurious actuations treated in the PRA? (B) Note 8 (NEDO-33201, page 2.2-18) states that the small and medium size steam LOCAs are consolidated and “the most restrictive treatment of the two categories is used in the PRA analysis.” Please provide more details about what was done,

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			including assumptions and associated bases. Also, please explain the basis of a related statement in NEDO-33201, Section 3.3.3.4 stating that "In this group of initiating events, it is considered that the size of the break does not affect the availability of any system that connects to the line containing the small LOCA." (C) Clarify why the following events from the operating BWR industry experience were either not considered or were re-classified: (1) very small LOCA/leak was not considered; (2) partial loss of condensate and feedwater (FW) was included in the "general transient" category which implies that the success criterion for FW is not accurate; (3) The spurious actuation of one or all four isolation condenser (IC) loops is mentioned but not anything in between; (D) Provide the basis for the assumption that the spurious actuation of one IC loop is bounded by the inadvertent opening of a relief valve (IORV). State whether the contribution of a spurious actuation of one IC loop is included in the assumed IORV frequency. (E) More detailed explanation of the categorization scheme and associated criteria regarding steam and liquid LOCAs is needed. It is stated that a break above L3 is classified as a steam LOCA, even though a liquid phase is initially discharged through the break, while a break below L3 is a liquid LOCA. Does this assumption result in a conservative analysis? Please explain and verify that this assumption is consistent with the initial water level used to determine PRA success criteria for safety systems and operator actions. (F) It is stated that malfunctions involving partial loss of the condensate or feedwater systems are included in the "General Transient" initiating event category. However, in the event tree for the General Transient category, the success criterion of one out of four is used. Please explain the validity of this success criterion. (G) It is stated (NEDO-33201, p. 2.2-12): "The loss of 13.8 or 6.9 kV AC buses of a single train produces a partial loss of plant service water system (PSWS), among other plant effects....[This event] it is not modeled as a separate initiator because the initiating event occurs only in case of failure to realign the redundant PSWS pumps (the frequency contributions of bus failure and operator error are judged not significant)." Please provide the basis of assuming that the frequency of a bus failure and operator

RAI Number	Reviewer	Question Summary	Full Text
			error (or unavailability due to maintenance) is not significant, especially in light of the fact that the consequences are similar to a station blackout event with additional plant failures. In addition, assumptions made in the human reliability analysis may lead to important design certification requirements even if it is shown that the loss of a 13.8 or 6.9kV AC bus is insignificant as an initiating event. (H) Provide the basis for statements that failure of a single PSWS train, one main steam isolation valve (MSIV) closure and a single feedwater or condensate train does not cause reactor trip. (Notes to NEDO-33201, Table 2.2-1)
19.1-24	Saltos N	Explain or clarify several assumptions made in quantifying the frequency of initiating event categories.	Explain or clarify assumptions in quantifying the frequency of initiating event categories. (A) NUREG/CR-5750 Event QG10, “Inadvertent Open/Close of one Safety/Relief Valve” is considered a contributor to both the frequency of general transients and IORV. Please explain. (B) It is stated that the large, medium and small break frequencies inside containment were calculated by apportioning proportionally the associated NUREG/CR-5750 frequencies to each group of ESBWR lines. However, it appears that these frequencies, as reported in NEDO-33201, Tables 2.3-2 and 2.3-3, are underestimated (e.g., the total large break frequency is about 2E-5/yr instead of 3E-5/yr reported in NUREG/CR-5750). Please explain. (C) The frequency of LOCAs outside containment resulting from main steam lines were obtained from NUREG/CR-4832 (specific to LaSalle Unit 2 Nuclear Power Plant). However, other sources (e.g., NUREG/CR-5750 and EPRI’s Utility Requirements Document) report significantly higher frequencies for main steam line breaks. Please explain why the information in these additional sources was not taken into account in estimating the frequency of ESBWR main steam line breaks. (D) The comparison of ESBWR PRA internal events initiating event frequencies to other studies, reported in NEDO-33201, Table 2.3-4, indicates a degree of variability that should be addressed with appropriate sensitivity studies. Please discuss.

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19.1-25	Saltos N	Clarify statement about the vessel rupture event not being maintained as an initiator category.	It is stated in NEDO-33201, Section 2.3.3.3 that vessel rupture is judged a negligible event and is not maintained as an initiator category for accident sequence quantification. However, an event tree for reactor vessel rupture is presented in Appendix A.3 (Figure A.3-20) and discussed in Section 3.3.3.5. Please clarify.
19.1-26	Saltos N	Provide more detailed information regarding interfacing LOCA.	It is stated in NEDO-33201, Section 2.3.3.4: "The interfacing LOCAs are of negligible frequency in the ESBWR design due to the numerous means of separation (i.e., check valves and air operated valves in the closed position, their opening interlocked with reactor pressure) between high and low pressure piping." The staff needs more detailed information that identifies important design and operational features of the ESBWR design which make the frequency of the interfacing LOCAs to be negligible. Please list all high-low pressure interfaces and for each such interface discuss how specific design and operational features (e.g., physical barriers, redundancy, diversity, pressure relief capability, interlocking, technical specifications (TS) or administrative controls) prevent interfacing LOCAs and justify the conclusion that the frequency of interfacing LOCAs is negligible. Please address conditions related to different possible plant configurations, system re-alignments and testing.

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19.1-27	Saltos N	The assumed mission time of 24 hours needs to be justified or extended for some accident sequences.	The assumed mission time of 24 hours (NEDO-33201, Section 3.2.5) may be inadequate for some accident sequences where the reactor coolant system conditions are not stabilized in 24 hours or core damage is anticipated following 24 hours without further system or operator action. For each of these accident sequences which are not assumed to end in core damage, the systems and/or operator actions needed to prevent core damage, need to be identified and their failure characterized and addressed in the PRA. If it cannot be shown that the residual risk is not significant (e.g., through a bounding analysis), the mission time should be extended (to a point in time when it can be argued that the residual risk is not significant). This information is of particular interest in the focused PRA (where no credit is given to the non-safety-related “defense-in-depth” systems to mitigate accidents) used in the regulatory treatment of non-safety systems (RTNSS) process. (A) Please discuss how these issues will be addressed and revise the PRA submittal accordingly. (B) It is mentioned in NEDO-33201, Section 3.2.5, that two sensitivity studies related to this issue are in Section 11. One study investigating the impact of not including CD II events (i.e., events where the containment fails while the core is successfully cooled for at least 72 hours) in the baseline core damage frequency (CDF), and the other investigating the impact of extending the mission time to 72 hours. The staff could not find these sensitivity studies in Section 11. Please clarify.
19.1-28	Saltos N	Discuss the basis, robustness and important assumptions associated with statements about containment failure and core cooling following containment failure.	Discuss the basis, robustness and important assumptions associated with the following statements related to containment failure and core cooling following containment failure: (A) In all cases utilizing passive systems, the containment does not fail until at least 24 hours following the initiating event, and any subsequent core uncovery does not occur until 72 hours following the initiating event. For example, address bounding conditions used and other conservative assumptions made in the analysis which provide confidence about the robustness of these two numerical values. (B) the containment failure does not affect continued indefinite core cooling when active water sources are used.

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19.1-29	Saltos N	Clarify success criterion for inventory requirement following loss of feedwater (event U2CISHORT).	It is stated that following a loss of feedwater event, the initial water level drop is much more severe than in a general transient. In order to avoid actuation of the Automatic Depressurization System (ADS), adequate supply of water inventory within a few minutes is required. The success criterion for this function (top event U2CISHORT) is stated as follows: "The inventory requirement only requires operation of the CRD [control rod drive] and IC condensate return lines for approximately 15 minutes to achieve success." Similar statements are made in the description of the loss of preferred power event tree. Please address the following: (A) Specify how many CRD pumps and how many IC loops are required for success and explain why only the IC condensate return lines, as opposed to IC loops, are required for success. (B) Provide the basis for the assumed 15 minutes time requirement (10 minutes for loss of preferred power) and explain how was this time modeled in the PRA (e.g., impact on the probability of avoiding ADS actuation). (C) How is this criterion changed in the focused PRA (where no credit is given to the non-safety-related "defense-in-depth" systems to mitigate accidents) used in the RTNSS process? (D) Clarify the heading descriptions for top events related to IC and CRD following success and failure of top event U2CISHORT and state the associated success criteria.
19.1-30	Saltos N	Clarify statement regarding ADS actuation and alternative means available for core cooling following a LOFW event.	Regarding the loss of feedwater (LOFW) event, it is stated that "If the water level drops below 1.5 but the ADS does not successfully actuate, ICS [isolation condenser system] remains available for core cooling. Additionally, if the ICS fails too, CRD and FAPCS [fuel and auxiliary pool cooling] in LPCI [low pressure coolant injection] mode (given depressurization using the SRVs only) could still provide core cooling." This statement seems to imply that failure of ADS actuation, as designed to occur, is actually a "success." If under these conditions the ICS can provide core cooling why does the design provide for an ADS logic actuation? Furthermore, if the accident can be mitigated without using ADS, what would prevent the operator from trying to inhibit automatic ADS actuation to avoid blowdown and its economic consequences? Please discuss.

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19.1-31	Saltos N	Clarify the modeling of station blackout and offsite power recovery in the loss of preferred power event tree.	The loss of preferred power event tree does not include top events for loss of diesel generators and recovery of offsite power (these failures are buried inside the fault trees developed for the loss of AC power). This approach makes it hard to understand how accident sequences involving station blackout and recovery of offsite power were modeled. It is stated that "...in case of diesel failure, there is the possibility of restoration of the offsite power before the reactor water level decreases below RPV [reactor pressure vessel] Level 1 to allow the use of the active systems given that ICS are ineffective." Please address the following: (A) What happens when RPV Level 1 is reached and why active systems cannot be used when the reactor water drops below RPV Level 1? (B) Does the statement refers to one or both diesels? (C) The assumptions made in calculating the probability to recover offsite power before RPV Level is reached. A more detailed discussion of the various event tree sequences, in terms of top events and success criteria (given the success or failure of preceding top events), is needed. Perhaps a separate event tree for station blackout and the split of top event U2CISHORT into two top events (for CRD and ICS) would make the loss of preferred power event tree more trackable. Please discuss.
19.1-32	Saltos N	Discuss the expected frequency of blowdown as modeled in both the baseline and focused PRAs.	In events where water level drops abruptly, such as loss of feedwater and loss of preferred power (loss of offsite power) transients, active high-pressure makeup is needed to prevent reaching the ADS actuation setpoint (see event U2CIDHORT). As modeled in the event trees, it takes the single failure of a non-safety-related system, such as a CRD pump or a diesel generator, to reach the ADS actuation setpoint. This implies that the blowdown frequency of ESBWR may be relatively high, especially in the focused PRA, given the frequencies of events such as loss of feedwater (about 1E-1 events/year) and loss of preferred power (about 5E-2/year) are not remote. Please provide the total expected frequency of blowdown as modeled in both the baseline and focused PRAs. This information is needed to assess the relative importance of active "defense-in-depth" systems to mitigate accidents and determine whether any requirements are needed to ensure availability and reliability commensurate with their importance.

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19.1-33	Saltos N	Clarify success criterion for top event XS5 (At least 5 SRVs open)	If no high pressure injection system is available, following a transient or small LOCA initiating event, it is necessary to depressurize (partially) by opening SRVs to permit effective injection using the FACPS or the Fire Protection System (FPS) pump. In Chapter 3 of the PRA (page 3.3-4) it is stated that the success criterion for this function is the manual opening of at least five of eight SRVs. However, in page 4.1-3 (where the ADS is discussed) it is stated that 10 of the 18 SRVs have the capability of actuating in the ADS mode. Please clarify why the above mentioned success criteria refers to only eight SRVs.
19.1-34	Saltos N	Discuss the basis for assuming that under all combinations of ADS configurations either the ICS or GDCS can effectively provide core cooling.	In the loss of feedwater and loss of preferred power event trees, it is assumed that core cooling can be achieved using the IC following the actuation of up to three DPVs. Please provide the basis for this assumption and discuss whether there is any pressure interval associated with ADS configuration where neither IC nor gravity driven cooling system (GDCS) can provide effective core cooling.
19.1-35	Saltos N	Discuss possibility of adverse interaction between the functions of the Steam Suppression System and the PCCS.	The Steam Suppression System has vacuum breakers (VBs) that must open when during a LOCA the pressure in the wetwell (WW) is greater than the pressure in the drywell (DW) to protect containment integrity and prevent back-flooding of the suppression pool water into the DW. However, all VBs must close and be leak tight to maintain a small pressure differential between the DW and the WW required for passive containment cooling system (PCCS) effectiveness. It appears that a new feature of the ESBWR design is an option to command the VBs to seal, in order to maintain such differential pressure. Was the potential to compromise the vacuum breaking function, by commanding the valves closed, investigated? Please discuss when, during the various phases of a LOCA accident, the vacuum breaking function should be available and the potential to be compromised.
19.1-36	Saltos N	Clarify the end state for the top two sequences of the Small Liquid LOCA event trees.	The top two sequences of the Small Liquid LOCA event trees (NEDO-33201, Figures A.3-18 and A.3-19) are both shown as no core damage, whether Isolation Condenser succeeds or not. Please clarify.

RAI Number	Reviewer	Question Summary	Full Text
19.1-37	Saltos N	Provide information on the modeling of the reactor scram function in the PRA (event tree heading C).	The reactor scram function (event tree heading C) is discussed briefly in NEDO-33201, Section 3. It is stated (page 3.3-10) that "The initiating event for the ATWS [anticipate transient without scram] event tree is a fault tree with the general transient ANDed with the failure of the RPS [reactor protection system] and/or ARI [alternate rod insertion]....." However, additional information on the modeling of this function is needed. Please address the following: (A) Provide the fault tree mentioned in the above statement (top event C71-SYS-FF-SCRAM) with adequate explanation to understand how the failure of the reactor scram function was modeled in the PRA. (B) Explain how the assumed probability of 5.8E-7 (reported in Chapter 11, Table 11-2) for failure to scram was calculated.
19.1-38	Saltos N	Provide information on the modeling of several top events	Provide information on the modeling of several top events, such as the overpressure protection function (event tree headings M and MA), the SRV re-closure in ATWS (event tree heading PA), the containment venting function (event T11-SYS-FF-OPEN), and the high pressure injection (event UCF) in ATWS. (A) The staff could not find events B21-SYS-FF-18/18SRV (1 of 18 SRVs must open), B21-SYS-FF-10/18SRV (9 of 18 SRVs must open), and B21-SYS-FF-1/9OPEN (all SRVs must re-close) in NEDO-33021, Table 4.1-8 (List of system top events for ADS) or the assumed probabilities for these events. Similar information on the modeling of the containment vent function failure is needed. (B) In Table 5.4-1 (Special Events), event T11-SYS-FF-OPEN is described as "all overpressure protection valves fail to open" and a probability of 5.69E-2 was assumed based on "Bounding Value, Engineering Judgment." Please explain. (C) Information on the modeling of the high pressure injection (event UCF) in ATWS is needed. Please clarify what top fault tree(s) were used for UCF since some ATWS events involve feedwater (FDW) run-back.

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19.1-39	Saltos N	Clarify the description of event heading WL (suppression pool cooling without depressurization).	The following statements are made in NEDO-33201, Section 3.3.2: "The decay heat removal function preferred for ATWS is accomplished by FAPCS in suppression pool cooling mode, if FAPCS is not previously actuated," and "This operational mode of FAPCS initiates automatically upon high temperature in the suppression pool, and if the FAPCS in the injection mode is not previously actuated." These two statements appear to imply that the FAPCS could have actuated previously in the injection mode, which conflicts with the ATWS event tree modeling (e.g., ADS inhibition). Please clarify.
19.1-40	Saltos N	Clarify the modeling of the failure to override the RWCU/SDC isolation signal following an ATWS event (event heading WHA).	In the ATWS event tree drawings, the top event for the decay heat removal function using the reactor water cleanup/shutdown cooling (RWCU/SDC) system, is shown as GG31TOP. However, this top event is not listed in Table 4.4-8 (List of system top events for RWCU/SDC). Please clarify how the operator failure to inhibit RWCU/SDC isolation signals, developed by ATWS logic, and bypass the filters was modeled in the PRA.
19.1-41	Saltos N	Clarify definition of "ATWS after Small LOCA above Core."	It is not clear what small LOCA categories are included in the "ATWS after Small LOCA above Core" event. Please clarify how "small LOCA above core" relates to level position (i.e., RPV Level 3).

Requests for Additional Information (RAIs)
ESBWR Design Control Document (DCD) Section 19.2.2.4

RAI Number	Reviewer	Question Summary	Full Text
19.5-3	Bagchi G	Update DCD Section 19.2.2.4 to include seismic margins analysis. (19.2.2.4)	DCD Tier 2, Section 19.2.2.4 provides a seismic margin analysis result of 0.6g for the High Confidence Low Probability of Failure (HCLPF). A seismic margins analysis to determine that the plant HCLPF for a certified design should be at least equal to 1.67 times the safe shutdown earthquake (SSE), based on criteria in SECY 93-087, “Policy, Technical and Licensing issues Pertaining to Evolutionary and Advances Light-Water Reactor (ALWRs) Designs,” April 2, 1993. The seismic margins analysis addressing the criteria in SECY 93-087 should be located in this section of the DCD. The associated structural calculations and assumptions need to be presented in DCD Tier 2, Chapter 19, showing all relevant assessments of the critical elements necessary to maintain plant performance during and after the SSE. References applicable to HCLPF calculations should be presented in Chapter 19.
19.5-4	Bagchi G	Update DCD Section 19.2.2.4 to include certified design components important for plant HCLPF analysis. (19.2.2.4)	All the certified design components important for the plant HCLPF analysis should be presented in a tabluar form in the DCD Tier 2, Chapter 19. Also, the table of HCLPF values in the ESBWR Probabilistic Risk Assessment (PRA) Report (NEDO-33201) should be incorporated into Tier 1 of the DCD as a part of an ITAAC item to ensure and verify that the as-built plant HCLPF is equal to or greater than the certified plant HCLPF value.
19.5-5	Bagchi G	Provide procurement specifications. (19.2.2.4)	Provide the essential elements of a procurement specification and associated installation criteria that would ensure that Structures, Systems and Components (SSCs) are procured and installed to develop the necessary HCLPF capacities.

RAI Number	Reviewer	Question Summary	Full Text
19.5-6	Bagchi G	Discuss functionality limits for drywell and wetwell and integrity of components attached to the reactor vessel. (19.2.2.4)	In Section 15.3.3 of NEDO-33201, Rev. 1, it has been recognized that relative displacements limiting SSC operability frequently control their seismic capacity. The structural fragility assessment method in Reference 15-1, R.P. Kennedy, et al., "Assessment of Seismic Margin Calculation Methods", NUREG/CR-5270, Lawrence Livermore National Laboratory, March 1989, is somewhat dated, and is based on a PWR plant study. The ESBWR design is very different—it has a very tall reactor vessel and drywell functionality is very much dependent on proper functioning of all pressure suppression components. Simply because of the reactor vessel height, a small amount rotation at the pedestal would significantly scale up the displacement near the reactor vessel head and the top of the drywell. Please discuss individual elements of functionality limits for ensuring drywell and wetwell functionality and the integrity of components attached to the reactor vessel.
19.5-7	Bagchi G	Provide failure modes for containment. (19.2.2.4)	Provide a description of the failure modes used to determine the HCLPF values for category I structures, particularly the containment structure. Provide a description of the extrapolation process supplemented by judgement.
19.5-8	Bagchi G	Describe how HCLPF values are determined for equipment and components qualified by testing. (19.2.2.4)	Provide a description of how HCLPF values are determined for equipment and components qualified by testing, especially for the North Anna early site permit (ESP) site-specific ground motion spectrum.
19.5-9	Bagchi G	Justify the use of both ductility and damping effects. (19.2.2.4)	Justify the use of both ductility (inelastic energy absorption factor) as well as damping (structural response factor) effects to determine the overall factor of safety.
19.5-10	Munson C	Provide a description of how generic fragilities were selected. (19.2.2.4)	Section 15.3.1 of NEDO-33201, Rev. 1, states that generic fragilities were chosen based on a review of prior PRAs and fragility data and that they are considered achievable for the ESBWRs with an evolutionary improvement in the seismic capacities of the components designed to a 0.3g SSE minimum. Provide a list of the prior PRAs and the bases for using their fragility values. If multiple fragility values for similar components were available, please describe the bases for the chosen value. Please describe where and how these generic fragility data were used to establish 0.6g HCLPF value for the ESBWR. Elaborate on the meaning of the phrase "evolutionary improvement" and how this ensures that these fragilities are achievable.

RAI Number	Reviewer	Question Summary	Full Text
19.5-11	Munson C	Clarify which PGA value was used in analyses. (19.2.2.4)	Section 15.3.1 of NEDO-33201, Rev. 1, states that the peak ground acceleration (PGA) of the design earthquakes is 0.3g for the SSE while the North Anna specific SSE has a PGA value of 0.49g. Please clarify which PGA value was used in your analyses to compute the capacity factors, particularly the strength factor (Fs). A certified design for the North Anna ESP response spectra would put the plant HCLPF value at 1.67x0.49g or about 0.82g, please explain how you meet the HCLPF criteria.
19.5-12	Munson C	Justify equation used to determine ultimate shear strength. (19.2.2.4)	Justify the use of Equation 15.3-11 in NEDO-33201, Rev. 1, to determine the ultimate shear strength for short reinforced concrete shear walls, typical of nuclear power plants. Provide the equation used to determine the ultimate shear strength for the containment wall.
19.5-13	Munson C	Justify use of NUREG/CR-0098 spectrum for input ground motion. (19.2.2.4)	For the shape factor (Fsa), Section 15.3.3.1.2 of NEDO-33201, Rev. 1, states that for the purpose of seismic risk assessment, the median ground motion spectrum given in NUREG/CR-0098, “Development of Criteria for Seismic Review of Selected Nuclear Power Plants,” is considered to be the realistic input ground motion definition. Considering the significant number of advancements in the field of seismic hazards since the development of this spectrum in the late 1970's, justify your consideration of the NUREG/CR-0098 spectrum as realistic input ground motion.
19.5-14	Bagchi G	Provide a comparison showing ratios of the bounding (all site conditions) seismic responses of the containment structure at important locations to the critical functionality limits. (19.2.2.4)	Provide a comparison showing ratios of the bounding (all site conditions) seismic responses of the containment structure at important locations to the critical functionality limits. Using the highest ratio determine the HCLPF value.

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