October 11, 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. 69 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.

RAI questions 15.0-16 and 15.0-17 relate to Chapter 15, "Safety Analyses," of the ESBWR design control document (DCD), Tier 2, Revision 1. These questions were sent to you in draft form via electronic mail on September 18, 2006, and discussed with your staff in a telecon on October 2, 2006. You agreed to respond to these RAI questions on November 22, 2006.

RAI questions 15.2-5 through 15.2-13, relate to analysis of anticipated operational occurrences (AOOs), as discussed in the ESBWR DCD, Tier 2, Revision 1, Chapter 15, "Safety Analyses." These questions were emailed to you in draft form on September 15, 2006, and discussed in a telecon with your staff on September 29, 2006. You agreed to respond to RAI questions 15.2-6 through 15.2-9, and 15.2-11 on November 9, 2006. You agreed to respond to RAI questions 15.2-5, 15.2-10, 15.2-12, and 15.2-13 on November 22, 2006.

RAI questions 15.3-4 through 15.3-24 relate to analysis of infrequent events, as discussed in the ESBWR DCD, Tier 2, Revision 1, Chapter 15, "Safety Analyses." To support the review schedule, you are requested to respond to this RAI set by November 22, 2006.

RAI questions 16.2-81 through 16.2-89 relate to the ESBWR DCD, Tier 2, Revision 1, Chapter 16, "Technical Specifications." These questions were sent to you in draft form via electronic mail on September 5, 2006. You did not request a telecon and agreed to respond to these RAI questions on November 22, 2006.

The staff redrafted RAI question 21.6-51 enclosed in this letter and is withdrawing RAI question 21.6-51 previously sent in RAI Letter No. 31 (ADAMS No. ML061740023). No response is needed for RAI question 21.6-51 provided in Letter No. 31. New RAI questions 21.6-92, 15.3-25, and 15.4-1 through 15.4-5 are enclosed in this letter. To support the review schedule, you are requested to respond to this RAI set by November 22, 2006.

If you have any questions or comments concerning this matter, you may contact me at (301) 415-4115 or mcb@nrc.gov or you may contact Amy Cubbage at (301) 415-2875 or aec@nrc.gov.

Sincerely,

/RA/

Martha C. Barillas, 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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ACCESSION NO. ML062830003

OFFICE	NESB/PM	NESB/BC
NAME	MBarillas	JColaccino
DATE	10/10/2006	10/11/2006

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Distribution for DCD RAI Letter No. 69 dated October 11, 2006

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Request for Additional Information (RAI)
ESBWR Design Control Document (DCD), Tier 2, Rev. 1, Chapter 15, Safety Analyses

RAI Number	Reviewer	Question Summary	Full Text
15.0-16	Huang T Thomas G Lois L	Provide the SLMCPR value in Chapter 15 of the DCD and in TS 2.1.1.2.	The safety limit for minimum critical power (SLMCPR) is a safety limit which is required to be specified in technical specifications (TS) according to 10 CFR 50.36(c)(1)(i)(A). The SLMCPR is the primary parameter for specified acceptable fuel design limits (SAFDL). Generic Letter (GL) 88-16 guidance specifies that core operating limits shall be established and documented in the core operating limits report before each reload or any remaining part of a reload cycle. It only applies to the core operating limits, not the safety limits. The operating limit minimum critical power ratio (OLMCPR) value is established based on the cornerstone of SLMCPR. The ESBWR DCD Tier 2, Chapter 15, Rev. 1, does not specify the SLMCPR value. The proposed TS section 2.1.1.2 does not specify the SLMCPR value either. Instead, TS 2.1.1.2 states, "Greater than 99.9 percent of the fuel rods in the core would be expected to avoid boiling transition." The proposed TS 2.1.1.2, uses a criterion instead of a specified SLMCPR value, and is not acceptable, as currently drafted. Please include the SLMCPR value in the DCD Tier 2, Chapter 15 for the equilibrium core assumed in the transient and accident analyses. Also, revise the proposed TS 2.1.1.2 to specify a SLMCPR value (e.g. 1.12 or other conservative value based on the preliminary analysis).

RAI Number	Reviewer	Question Summary	Full Text
15.0-17	Thomas G Lois L Clifford P	Acceptance Criteria for Infrequent Events	Standard Review Plan (SRP) Section 15.2.8.II.A.1, Revision 1, July 1981, states that pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures [ASME Boiler and Pressure Vessel Code, Section III, Service Level B] for low probability events and below 120 percent of the design pressures [ASME Boiler and Pressure Vessel Code, Section III, Service Level C] for very low probability events such as double-ended guillotine breaks. DCD Tier 2, Rev. 1, Table 15.0-5, Acceptance Criteria for Infrequent Events, states that "pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120 percent of design pressure." Revise DCD Tier 2, Table 15.0-5 consistent with the SRP acceptance criteria for low probability events that the pressure in the reactor coolant and main steam systems should be maintained below 110 percent (Service Level B) of the design pressures.

RAI Number	Reviewer	Question Summary	Full Text
15.2-5	Lois L	Discuss consequences of partial failure of SCRRI	DCD Tier 2, Rev. 1, Figure 15.2-1e demonstrates the importance of the selected control rod run-in (SCRRI) insertion for mitigation of this transient. The ESBWR is physically a very large core. If a partial failure of SCRRI were to occur, how would ESBWR avoid violating local thermal limits or creating a core instability without shutting down the core? See DCD Tier 2, Figure 15.2-1a.
15.2-6	Lois L	Discuss mechanical failure of SB&PC triplicated control system	DCD Tier 2, Rev. 1, Section 15.2.2 and elsewhere, states that the Steam bypass and pressure control (SB&PC) triplicated control system is not subject to a credible single failure. The statement seems to be based on circuitry and electronic operation of the system alone. However, mechanically the system is mounted on a single device(s). Sections 15.2 and 15.3 emphasize the electronic control systems but ignore the possible contribution of the associated mechanical systems. For example, steam bypass or safety relief valves or control rods do not seem to contribute to the transient frequencies involving these components. A number of transients that could be initiated by SB&PC failure are in the anticipated operational occurrence (AOO) category. Have you included mechanical failure of the device(s)? Please give the estimated failure frequencies of the triplicate control system. Are the electronic components themselves free of failures due to mechanical, heating, testing, vibration, and other causes?
15.2-7	Lois L	Address TMI Action Item II.K.3.16	Per Standard Review Plan (SRP) Section 15.6.1, Revision 1, July 1981, TMI Action Item II.K.3.16, evaluation of the safety relief valve (SRV) performance, should be addressed in the DCD and the results should be included in the frequency evaluation and categorization of the inadvertent opening of an SRV event. Has this issue been addressed in the ESBWR DCD?

RAI Number	Reviewer	Question Summary	Full Text
15.2-8	Lois L	Address mechanical failures of bypass and turbine control valves failures	In DCD Tier 2, Rev. 1, Section 15.2 (and elsewhere), the transients call for bypass and turbine control valves to open and close. Instrumentation failures are considered, but valve mechanical failure is not. Examples include failure to reseat or stuck closed valves, for which there exists a considerable database of experience. Address the issue of valve mechanical failures in the context of creating a non-analyzed condition or a new transient.
15.2-9	Lois L	Explain assumed turbine stop valve closing time	DCD Tier 2, Rev. 1, Tables 15.2-10 and 15.2-11 show turbine stop valve closing times of 0.16 and 0.10 seconds respectively. Conditions seem to be identical and the 0.16 is designated as “realistic closure timing.” Provide explanation for using 0.10 seconds in DCD Tier 2, Table 15.2-11 and whether this is a realistic closure time. Explain the effect on the transient when a realistic closure time value is used.

RAI Number	Reviewer	Question Summary	Full Text
15.2-10	Lois L	Address reactivity anomalies resulting from mechanical failure of control rods	DCD Tier 2, Chapter 15 dismisses reactivity anomalies in the AOO category. The DCD does not include any justification that the control rod malfunctions are common (within or greater frequency than 10e-2). The argument has been made that the electronic portion of the control system has been improved. However, the mechanical part of control rod insertion/withdrawal is not mentioned. The Appendix 15A "Event Frequency Determination," sections 15A.3.11-13 regarding control rod errors during refueling, startup and operation, finds that inadvertent criticality to be at most 1.0e-7, 1.2e-6 and 1.5e-7 per RY, respectively. Such frequencies would qualify to be analyzed in the infrequent events or in the accidents section. Section 15.3.7.2 states that "During refueling....interlocks provide assurance that inadvertent criticality does not occur..." Likewise, section 15.3.9.2 concludes that "There is no basis for occurrence of the continuous control rod withdrawal error event in the power range." Yet, the probability estimates are in the same range as in the startup case for which some kind of analysis was provided. Justify the exclusion of reactivity anomalies from AOOs and include the mechanical part of the reliability including test data of the ESBWR control rod system. Was the difference in the estimated probability values of reactivity transients for refueling and power operation versus the startup the reason not to analyze refueling and power operation reactivity transients? If operational data was used in the estimation of the probability of reactivity transients, please describe the data used. Are the electronic components themselves free of failures due to mechanical, heating, testing, vibration, and other causes?

RAI Number	Reviewer	Question Summary	Full Text
15.2-11	Lois L	Explain control rod reactivity insertion during inadvertent IC initiation event	The calculated results of the transient resulting from inadvertent isolation condenser (IC) initiation are shown in DCD Tier 2, Rev. 1, Figure 15.2-11. In this figure (as well as Figure 15.2-1, depicting a similar transient) positive control reactivity is inserted at the same time as reactor power is increasing. In both instances, (but mainly in Figure 15.2-1) the minimum critical power ration (MCPR) gets close to or lower than 1.30. This action appears counter intuitive and appears to be the wrong thing to do. Explain why the system is designed to insert reactivity at that particular time.
15.2-12	Lois L	Explain FW pump runout event	The term “pump runout” implies excessive pump flow into lower pressure than the design pressure. DCD Tier 2, Rev. 1, Table 5.2-19 states “at system design pressure.” Explain how the FW pump is able to increase its flow against design pressure for the Runout of One Feedwater Pump event.
15.2-13	Lois L	Address FW pump runout categorization	DCD Tier 2, Rev. 1, Section 15.2.4.2 includes extensive discussion of the improved electronics and conveys the impression that pump runout is a very low probability event. If this is the case, why is pump runout categorized in DCD Section 15.2 rather than Section 15.3?

RAI Number	Reviewer	Question Summary	Full Text
15.3-4	Dehmel JC	Provide dose results and supporting information for the accident scenario involving the failure of a tank containing liquid radioactive wastes.	A review of DCD Rev. 1, Tier 2, Section 15.3.16 indicates that dose results are not provided for the analysis assessing the failure of a tank containing liquid radioactive wastes. Address the following inconsistencies with NRC guidance and acceptance criteria; describe the method, basis and assumptions used in the analysis; provide a listing of parameters used in the analysis; update the text in DCD Section 15.3.16; revise all supporting tables in DCD Section 15.3.16; and revise DCD Section 15.0 and Tables 15.0-1 to 15.0-7, as needed: a. DCD Section 15.3.16.3 states that dose results are presented in Table 15.3-14, but this table presents only noble gas release rates for a scenario involving offgasing from 1000 failed fuel rods. b. DCD Section 15.3.16.3 states that parameters are listed in Tables 15.3-12 and 15.3-13. However, Table 15.3-12 deals with an event for stuck safety relief valves, and Table 15.3-13 presents parameters for a scenario involving the failure of 1000 fuel rods. c. DCD Section 15.3.16.3 states that the iodine inventories are based on DCD Section 12.2, but it does not specify which type of liquid wastes was selected out DCD Section 12.2. d. DCD Section 15.3.16 does not acknowledge the criteria and guidance of SRP Section 15.7.3. Specifically, 10 CFR Part 20 Appendix B effluent concentration limits, GDC 60, and basis for radioactivity inventory based on a specific offgas release rate and delay time for BWRs. e. DCD Section 15.3.16 does not identify the analytical method used, i.e., methodology of NUREG-0016, NUREG-0133 (App. B for BWR), or other unspecified approach.

RAI Number	Reviewer	Question Summary	Full Text
15.3-5	Dehmel JC	A review of DCD Rev. 1, Tier 2, Section 15.3.16 indicates that the technical approach is not consistent with that described in SRP (NUREG-0800) Sections 15.7.3. II and 15.7.3.III.	DCD Rev. 1, Tier 2, Section 15.3.16 indicates that the technical approach is not consistent with that described in SRP Sections 15.7.3. II and 15.7.3.III. The analysis considers only a single pathway involving airborne releases of radioactivity via the HVAC system. This approach takes credit for the presence of a liner designed to contain the volume of the tank into the compartment where the tank is assumed to be located. However, the text states that this design feature applies only to tanks containing "high level liquid radwaste." The implication is that tanks containing low level liquid radwaste would not be located in compartments that afford the same level of protection. Moreover, Sections 11.2 and 11.4 of the DCD emphasize the use of mobile liquid and wet-waste processing systems. Given this design, discuss if the analysis considers failure of tanks that are part portable waste treatment systems and whether the placement of portable radwaste processing systems are afforded the same level of protection as that to permanently installed tanks. Provide the basis as to why the stated approach is consistent with SRP Section 15.7.3.III, which states that: "Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks. Credit is not allowed for retention by coatings or leakage barriers outside of the building foundations." Address these inconsistencies with NRC guidance and acceptance criteria; describe the basis and assumptions used in the analysis; discuss why the release of a tank's content to surface or groundwater is not limiting in the analysis; update the text in DCD Section 15.3.16; and provide new or revise all supporting tables in DCD Section 15.3.16.

RAI Number	Reviewer	Question Summary	Full Text
15.3-6	Lois L	Provide reactor shutdown signal beyond SCRRI.	DCD Tier 2, Rev. 1, Figure 15.3-1 indicates that the plant stabilizes at 120% power, 105 percent FW flow and 1.24 MCPR. Under those conditions the plant keeps increasing fuel damage. Should the plant be scrammed by other means than the Selected Control Rod Run-In (SCRRI)?
15.3-7	Lois L	Quantify uncertainties justifying assumed SCRRI failure.	DCD Tier 2, Rev. 1, Section 15.3.1.1, states "...SCRRI is assumed to fail and reactor scram on high simulated thermal power is not credited due to uncertainties. Therefore, a new steady state is reached." In view of your position in 15.2 that the SCRRI and the control rod system was not assumed to fail, please quantify the uncertainties that justify the particular mechanism leading to the assumed SCRRI failure.
15.3-8	Lois L	State the different FWCS controls for the different operating modes.	The FWCS is described in section 7.7.3.2 of the DCD, Tier 2, Rev. 1,and states that for power levels ≤ 25 percent, it uses single element control with regards to operating modes. Does single element control provide the same degree of reliability as triple element operation? At normal power range operation the three-element control mode is utilized. Is there an "intermediate" (two element) control mode? What is the interface power level?
15.3-9	Lois L	State the rods in transition boiling.	Provide the actual number of rods in transition boiling in DCD Tier 2, Rev. 1, Section 15.3.1. Is the number calculated assuming the MCPR shown in Fig. 15.3-1g or is the value attained if the high-power scram operated?
15.3-10	Lois L	Provide the basis for the rod failures with MCPR > 1.30.	DCD Tier 2, Rev. 1, Section 15.3.4.4 states "...there are no fuel rods that enter transition boiling (MCPR > 1.30)." That is also indicated in Figure 15.3-4g. What is the meaning of Sections 15.3.4.5 and 15.3.1.5 analyzing 1000 failed rods, and what is the basis for the numerical value chosen?

RAI Number	Reviewer	RAI Summary	Text
15.3-11	Lois L	Explain 15.3.5 transient duration.	For DCD Tier 2, Rev. 1, Figures 15.3-5a and 15.3-5g, has TRACG been qualified and benchmarked to calculate the narrow sharp power peak shown in Fig. 15.3-5a? For the power to increase, voids should collapse from the pressure wave created by closing the TCVs. The power peak full-width at half-maximum is less than .25 seconds. Does this represent physical reality?
15.3-12	Lois L	Provide 15.3.6 transient calculation basis.	Please consider the same questions as in RAI 15.3-10 above for the event covered in section 15.3.6 of DCD Tier 2, Rev. 1.
15.3-13	Lois L	Explain the control rod withdrawal at startup.	DCD Tier 2, Rev. 1, Section 15.3.8.1 states that "...the SRNM has a period based trip function ... by initiating a rod block ... for a period shorter than 20 seconds." Section 15.3.8.3.2 states that "... the control rod withdrawal speed is 30 mm/s, the nominal .... speed..." The analysis results show that the rod assembly (gang) withdrawal generates a period of about 4 seconds. Does this mean that normal startup creates a 4 second period liable to rod block (this section description seems to be incomplete)?
15.3-14	Lois L	State failures considered in Sections 15.3.2/3/4.	For the events in DCD Tier 2, Rev. 1, Sections 15.3.2/3/4, the probability of failure seems to be based on the improved triplicate logic of the SB&PC system. However, operating experience shows that most of the high pressure valve failures do not originate with the electronic logic, but the mechanical functions of the valve, i.e., failure to close/open or sticking half open. Have such failures been accounted for in the calculation of the probabilities of these events for their categorization?

RAI Number	Reviewer	RAI Summary	Text
15.3-15	Lois L	Explain CRD operation and provide the equilibrium water level for the event discussed in Section 15.3.4.	DCD Tier 2, Rev. 1, Section 15.3.4, Figure 15.3-4a indicates that the reactor shuts down at about 2 seconds, the FW flow is at a minimum at 5 seconds, and keeps increasing at 20 seconds as vessel pressure decreases. The sequence table calls for long term operation of the high pressure CRD pumps. It seems that long term, the vessel will have too much water, rather than too little. Explain why you call for high pressure CRD operation and what will be the anticipated eventual equilibrium vessel water level. State the reason for not mentioning IC initiation.
15.3-16	Lois L	Provide basis for determination of probability of bypass failure.	DCD Tier 2, Rev. 1, Section 15.3.5, at the end of paragraph 15.3.5.1, you divide the actuarial probability for bypass failure with load rejection by 100 to calculate the failure probability. This factor is attributed to the triplicate electronic control system. In Sections 15.3.1, 15.3.3 and elsewhere, valve failure probability was based on the electronic portion of the control and ignored the mechanical aspects of valve failure. Explain the omission of the mechanical aspects and the difference that justifies your choice.
15.3-17	Lois L	Explain water level decrease and FW flow increase for the event discussed in Section 15.3.5.	In DCD Tier 2, Rev. 1, Figure 15.3-5a, explain the water level decrease as steam flow stops, FW flow continues, the SRVs do not open, and the IC is not yet operational. Explain initiation of the CRD high pressure injection at the end of the calculated part of the transient as the FW flow keeps increasing.
15.3-18	Lois L	State signal and action to prevent steam pipe flooding for the event in Section 15.3.6.	In DCD Tier 2, Rev. 1, Section 15.3.6, the water level decreases for no apparent reason. While FW flow keeps increasing and the HP CRD injection is activated, what signal and what action will prevent the steam pipe from flooding?

RAI Number	Reviewer	RAI Summary	Text
15.3-19	Lois L	Provide the basis and analysis justifying the conclusion for the event in Section 15.3.7.	DCD Tier 2, Rev. 1, Section 15.3.7 states rod withdrawal error during refueling is characterized as impossible due to interlocks and design improvements. The staff finds this position unacceptable because it refers to an untested design. There is no detailed discussion for this conclusion provided in the DCD. Provide the basis for reaching this conclusion and the analysis demonstrating the magnitude of the consequences for this event under refueling conditions.
15.3-20	Lois L	Justify not analyzing the malfunctions of the automated rod movement control system leading to inadvertent reactivity transients.	DCD Tier 2, Rev. 1, Section 15.2 states that no inadvertent reactivity transients could be found. Section 15.3.8.1 of the DCD states that reactivity transients can be caused by "...malfunctions of the automated rod movement control system." Section 15.3.9 of the DCD states that "There is no basis for occurrence of the continuous control rod withdrawal error event in the power range." Malfunctions are not controllable, thus, they could be part of the AOOs and/or any power level of operation and should be analyzed accordingly. Provide the basis for not analyzing the malfunctions leading to inadvertent reactivity transients and the inconsistency in the referenced sections.
15.3-21	Lois L	Justify not analyzing the rod withdrawal error event.	DCD Tier 2, Rev. 1, Sections 15.3.9.2/3 states that the plant design precludes the rod withdrawal error event from happening and therefore, there is no need to analyze this event. The description in the DCD does not allow the reviewer to conclude that the event is impossible and thus, an analysis is not needed, since the analysis is the means to decide whether the design is safe. Provide the justification to forgo the analysis and include discussion of the electronic and the mechanical aspects of the design in the justification. Justify that the Automated Thermal Limit Monitor system will never fail and the possibility of not removing the permissive for rod withdrawal. Provide the basis for the probability of the mechanical system failure used in your justification.

RAI Number	Reviewer	RAI Summary	Text
15.3-22	Lois L	Justify classification of event and provide temperature and reactivity changes associated with the event discussed in Section 15.3.12.	DCD Tier 2, Rev. 1, Section 15.3.12.1 states that the power level will settle at a new steady state without violating the thermal limits. Section 15.3.12.2 states that the operator may take action to limit the power rise. Flux scram occurs if no operator action is taken. Therefore, for the thermal limits not to be violated either a scram or an operator action is required. Thus, the statement in Section 15.3.12.1 is not correct. If there is no additional failure why is this transient in the infrequent events? (1) Quantify the temperature and reactivity changes. (2) Provide the reason why this transient should not be classified as an AOO.
15.3-23	Lois L	Provide the basis for categorizing the inadvertent opening of an SRV in Section 15.3.13 as an IE.	In existing power reactors, operating experience shows there have been several inadvertent SRV openings and particularly incidents of partial closure. For DCD Tier 2, Rev. 1, Section 15.3.13, how did GE figure that the probability for this occurrence is in the infrequent event category? Was the mechanical history of SRV performance accounted for? What are the mechanical/ electronic (signal) improvements and associated databases to justify this categorization.

RAI Number	Reviewer	RAI Summary	Text
15.3-24	Thomas G	Provide a more detailed discussion regarding the rad. assessments for IE.	In DCD Tier 2, Rev. 1, Chapter 15, Table 15.3-1, the calculated delta CPR for the Infrequent Events varies from 0.0 to 0.15. Since the calculated delta CPR for all the events are small, according to the table, SLMCPR is not violated. However, radiological assessments are provided for several of them. Provide a more detailed discussion regarding the radiological assessment in spite of the low delta CPR. Also, specify the OLMCPR and the SLMCPR assumed in the analyses Subsection I.D given in the table is incorrect. Make the following editorial changes to the next DCD revision: 15.3.2.2 to 15.3.3 15.3.2.3 to 15.3.4 15.3.2.4 to 15.3.5 15.3.2.5 to 15.3.6 15.2.2.6 to 15.3.13 15.2.2.7 to 15.3.15

RAI Number	Reviewer	RAI Summary	Text
16.2-81	McConnell M Morris G	Describe how the proposed ESBWR TS SRs for determining the battery's state-of-health meet GDC 18.	General Design Criterion (GDC) 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing. The proposed ESBWR Class 1E batteries are to be designed with 24 and 72-hour duty cycles. Although not specified, the IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," was not developed with the notion of testing batteries with lengthy duty cycles (i.e., 24 and 72 hours). Describe how the proposed Surveillance Requirements (SRs), in DCD Chapter 16 TS, rev 1, for determining the battery's state-of-health (i.e., service testing, modified performance testing, and performance discharge testing) meet GDC 18.
16.2-82	McConnell M Morris G	Provide the basis for the proposed Completion Times for LCO 3.8.1 CONDITION B and LCO 3.8.5 CONDITION A.	Provide the basis for the proposed Completion Times in DCD Tier 2, Chapter 16, Rev. 1, TS Limiting Conditions for Operation (LCO) 3.8.1 CONDITION B and LCO 3.8.5 CONDITION A.
16.2-83	McConnell M Morris G	Provide the basis for the lack of a CONDITION for TS 3.8.2.	Provide the basis for the lack of a CONDITION for an inoperable 72-hour battery in the DCD Tier 2, Chapter 16, Rev. 1, proposed TS 3.8.2.
16.2-84	McConnell M Morris G	Provide the basis for the brackets around SRs 3.8.1.2 and 3.8.1.4 and around '≥rated' in SR 3.8.2.2.	Provide the basis for the brackets around SRs 3.8.1.2 and 3.8.1.4 and around '≥rated' in SR 3.8.2.2 in DCD Tier 2, Chapter 16, Rev. 1, TS.

RAI Number	Reviewer	RAI Summary	Text
16.2-85	McConnell M Morris G	Describe the 'alternate means' statement in the Bases section that is being crediting for this AOT.	DCD Tier 2, Chapter 16, Rev. 1, TS 3.8 states a 7-day allowed outage time (AOT) for Required Action A.3 of LCO 3.8.1 and LCO 3.8.2. Describe the 'alternate means' statement in the Bases section that is being crediting for this AOT.
16.2-86	McConnell M Morris G	Provide assurance that a battery with a battery pilot cell with a voltage of 2.07 volts or slightly greater will remain capable of performing its minimum designed function.	DCD Tier 2, Chapter 16, Rev. 1,TS 3.8 proposes a 2.07 volt limit was proposed when measuring the battery pilot cell voltage. The battery pilot cell is representative of the average battery cell in the battery. Provide assurance that a battery with a battery pilot cell with a voltage of 2.07 volts or slightly greater will remain capable of performing its minimum designed function (LCO 3.8.4 CONDITION A, LCO 3.8.4 CONDITION F, and SR 3.8.4.2).
16.2-87	McConnell M Morris G	Provide assurance that a battery with a battery pilot cell electrolyte temp. slightly greater than or equal to the min. established design limit will remain capable of performing its minimum designed function.	DCD Tier 2, Chapter 16, Rev. 1, TS 3.8 provides no actions for identifying or restoring the temperature of other battery cells that are above the minimum design limits. Provide assurance that a battery with a battery pilot cell electrolyte temperature slightly greater than or equal to the minimum established design limit will remain capable of performing its minimum designed function (LCO 3.8.4 CONDITION D and SR 3.8.4.4).
16.2-88	McConnell M Morris G	Provide the basis for not including over-current protection in the proposed ESBWR TS.	Section 8.3.1.4.1 of the DCD Tier 2, Rev. 1, under the heading "Electric penetration assembly," states that "redundant overcurrent interrupting devices are provided for all electrical circuits going through containment penetrations, if the maximum available fault current (including failure of upstream devices) is greater than the continuous rating of the penetration. This avoids penetration damage in the event of failure of any single over current device to clear a fault within the penetration or beyond it." Provide justification why these devices were not included in the TS in accordance with Criterion 3 of 10CFR 50.36(c)(2)(ii).

RAI Number	Reviewer	RAI Summary	Text
16.2-89	McConnell M Morris G	Provide justification for referencing IEEE 450-1995, and ensure that the battery maintenance program in proposed TS 5.5.10 is comprehensive	TS Section 5.5.10 contains a reference to the Institute of Electrical and Electronics Engineers (IEEE) Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications." The most recent version of IEEE Standard 450 that has been endorsed by the NRC through Regulatory Guides (RGs) is IEEE Standard 450-1975. The RGs of mention are: RG 1.128, "Installation, Design, and Installation of Large Lead Storage Batteries for Nuclear Power Plants," and RG 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants." a. Provide the justification for referencing IEEE Standard 450-1995. b. Provide assurance that all essential maintenance parameters have been included in battery monitoring and maintenance program identified in proposed TS 5.5.10.

Requests for Additional Information (RAIs)ESBWR Design Control Document (DCD) Chapter 21
Related to NEDE-33083P Supplement 2
"TRACG Application for ESBWR Anticipated Transient Without Scram Analysis"

RAI Number	**Reviewer**	**Question Summary**	**Full Text**
21.6-51	Klein V	Demonstrate ESBWR is stable during loss of feedwater and turbine trip events.	Evaluate whether or not instability is likely to occur during the following ATWS events: a) Loss of Feedwater Flow and b) Turbine trip with full bypass and feedwater available. Do not model any operator actions, but include the automated actions (e.g., feedwater runback on high pressure scram) if setpoints are reached. Using your approved methodology NEDE-33083P, Supplement 1 "TRACG Application for ESBWR Stability Analysis," determine a decay ratio beyond the scram setpoint, when power is raised beyond reactor scram setpoint for the Turbine trip event and level lowered beyond reactor scram setpoint for the Loss of Feedwater Flow event. Power and level should be justified for each of the events. Alternatively, add margin to your calculations by increasing the void reactivity coefficient by 30 percent.
21.6-92	Klein V Yarsky P	Provide code versions for all analyses in DCD Tier 2 Chapter 4, 6 and 15.	For each analysis performed in Chapters 4, 6 and 15, update the DCD Tier 2 to include the specific codes used including exact version, revision, and modification designations. In instances where a suite of codes is used (i.e., TRACG with a PANACEA wrap up file and GSTRM gap conductance model), include this information for each code used as part of the suite. Identify the software test report number associated with each production code.

RAI Number	Reviewer	Question Summary	Full Text
15.3-25	Lee J	Provide complete source term information for the radiological consequence analysis for certain infrequent events.	A review of DCD Rev. 1, Tier 2, Section 15.3 has not provided complete source term information for the radiological consequence analysis for certain infrequent events. In Section 15.3, you listed 16 infrequent events. Out of these16 infrequent events, you have performed and provided the radiological consequence analyses for the following six infrequent events: Section 15.3.1Loss of Feedwater Heating with Failure of Selected Control Rod-In Section 15.3.4Pressure Regulator Failure - Closure of all Turbine Control and Bypass Valves Section 15.3.5Generator Load Rejection with Total Turbine Bypass Failure Section 15.3.6Turbine Trip with Total Turbine Bypass Failure Section 15.3.10 Fuel Assembly Loading Error - Misloaded Bundle Section 15.3.11 Fuel Assembly Loading Error - Misoriented Bundle Please provide the following additional source term information for the staff to perform an independent confirmatory dose calculation for the infrequent events listed above: (A) Technical bases for assuming 1000 fuel rod failure with no fuel melt. (B) Complete fission product inventory in reactor core at 4590 Mwt power level and state methodology used for developing the core inventory of fission products. (C) Table 15.3-15 is titled as “1000 Fuel Rod Failure Core Fission Product Inventory.” Are these fission product inventory in this table represent total fission product inventory in only 1000 failed fuel rods? Have you applied the radial peaking factor to these values? (D) Condenser leak rate and duration of release from the condenser to the atmosphere. (E) Amount of fission products released to the primary coolant from 1000 failed fuel rods. (F) Amount of fission products reached the turbine and condenser.

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			(G) Amount of fission products released to the environment as function of time (for 0 to 2, 2 to 8, and 8 to 24 hours)): • through condenser • through off-gas system (H) Technical bases for offgas dynamic adsorption coefficients and xenon holdup time in absorber beds used in dose calculation. (I) Control room operator doses for these events (J) Radiological consequence dose calculations performed for the Exclusion Area boundary (EAB), Low Population Zone (LPZ), and Control Room (CR). If an NRC computer code was used for the dose calculation (i.e., RADTRAD), please provide its input and output files.
15.4-1	Lee J	Provide source term assumptions for Fuel Handling Accident	DCD Tier 2, Revision 1, Section 15.4.1, “Fuel Handling Accident,” describes the postulated fuel handling accident and provides the resulting radiological consequence analysis. This section needs additional information regarding source term assumptions used by the applicant in order for the staff to perform an independent radiological consequence analysis. (A) Please provide noble gases and iodine activity inventories in the fuel rod gaps (a) during normal operation at 4590 MWt with an average fuel burnup of 35 GWd/Mt, and (b) prior to fuel movement after 24 hour decay period that is available for release to the water surrounding the failed fuel assemblies. Also, please provide the amount of noble gases and iodine activities released to the environment following the postulated FHA. (B) In DCD Tier 2, Revision 1, Table 15.4-2, “FHA Parameters,” you provided specific values and parameters used in the postulated FHA analysis. Please provide technical bases for

RAI Number	Reviewer	Question Summary	Full Text
			assuming four failed fuel bundles due to the postulated FHA and state where this event is assumed to occur (i.e., inside containment, fuel handling area, auxiliary building, spent fuel pool). (C) Please provide atmospheric dispersion factors (χ/Q values) used for the EAB, LPZ, and CR. (D) Please include in DCD Tier 2, Table 15.4-2 (1) the fraction of fission product in fuel gap used, (2) depth of water in the spent fuel pool that available for scrubbing fission product fission products before it is released, and (3) the release points (pathways) from the plant to the environment. (E) If the FHA occurs in the containment, do you require closure of the containment purge lines? If you do require the closure, please state how you initiate the closure of the containment purge lines. If you rely on a radiation monitor to detect high airborne radioactivity, please state the sensitivity, range and setpoint of the radiation monitor. Is the ESBWR technical specifications require containment and/or Fuel Building closed during fuel movement, maintaining its integrity? (F) In DCD Tier 2, Revision 1, Section 15.4.1.2 lists seven items of "Identification of Operator Actions" to be carried out by operators following postulated FHA. Please state any of these actions are subjected in the radiological consequence analysis, ESBWR technical specifications, and/or COL Action Items. (G) The fuel and auxiliary pools cooling system for the spent fuel pool cooling and cleanup is a non-safety related system. Therefore, a loss of spent fuel pool (SFP) cooling capability should be analyzed coincident with the postulated FHA. The loss of SFP cooling could result in the pool reaching boiling, and a portion of the radioactive iodine in the SFP water could be released to the environment. Please provide the radiological consequence analysis for SFP boiling with a coincident loss of SFP cooling capability. (H) Please state if 24 hour decay time, prior to movement of irradiated fuel, assumed in the

RAI Number	Reviewer	Question Summary	Full Text
			FHA analysis is specified in the ESBWR Technical Specification as a limiting condition for operation (LCO). (I) Please provide complete FHA radiological consequence dose calculations performed for the EAB, LPZ, and CR. If an NRC computer code was used for the dose calculation (i.e., RADTRAD, HABIT), please provide its input and output files. (J) In DCD Tier 2, Revision 1, Section 15.4.1.4.2 lists "Assumptions to be Confirmed by the COL Applicant." Please state if these items will be specified as COL Action Items and/or Inspection, Test, Analysis and Acceptance Criteria (ITAAC) items. (K) Please provide basis for reactor building release rate assumed as 350 percent per day. (L) Please confirm the FHA isotopic release values to the environment provided in DCD Tier 2, Revision 1, Table 15.4-3, "FHA Isotopic Release to Environment," are correct.
15.4-2	Lee J	Provide source term assumptions for main steamline break accident Outside Containment	DCD Tier 2, Revision 1, Section 15.4.5, "Main Steamline Break Accident Outside Containment," describes the postulated a large steam line break outside containment and provides the resulting radiological consequence analysis. This section needs additional information regarding source term assumptions used by the applicant in order for the staff to perform an independent radiological consequence analysis. (A) Please state if the five second main steam isolation valve closure time, assumed in the radiological analysis, is specified in the ESBWR Technical Specification. (B) Please provide control room operator doses for both with an assumed pre-accident iodine spike and an accident-initiated iodine spike. (C) Please provide complete radiological consequence dose calculations performed for the EAB, LPZ and CR. If an NRC computer code was used for the dose calculation (i.e., RADTRAD, HABIT), please provide its input and output files.
15.4-3	Lee J	Provide source	DCD Tier 2, Revision 1, Section 15.4.8, "Failure of Small Line Carrying Primary Coolant

RAI Number	Reviewer	Question Summary	Full Text
		term assumptions for Failure of Small Line Carrying Primary Coolant Outside Containment	Outside Containment," describes the postulated a small steam or liquid line break inside or outside the containment and provides the resulting radiological consequence analysis. This section needs additional information regarding source term assumptions used by the applicant in order for the staff to perform an independent radiological consequence analysis. (A) You stated in DCD Tier 2, Revision 1, Section 15.4.8.5.1 that the SRP does not provide detailed guidance. The staff believes the detailed guidance is provided in SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment.," Revision 2, July 1981. Please state if you have taken any exceptions or deviations from the guidance provided in SRP Section 15.6.2. (B) Please provide steam/water break flow rate(s) used in your dose calculation. (C) Please provide a copy of dose calculation performed including determination of iodine appearance rates and resulting iodine concentrations due to the iodine spike. (D) You used reactor building flow (leak) rate of 200 percent per hour. Is this value in the ESBWR technical specification? 10 CFR 50.36(c)(ii)(C) criterions requires that a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier should be specified as a limiting condition for operation.
15.4-4	Lee J	Provide additional information regarding source term assumptions for RWCU/SDC line failure outside containment	DCD Tier 2, Revision 1, Section 15.4.9, "Reactor Water Cleanup/Shutdown Cooling [RWCU/SDC] System Line Failure Outside Containment," of the ESBWR DCD describes the postulated reactor water cleanup system line failure outside containment and provides the resulting radiological consequence analysis. This section needs additional information regarding source term assumptions used by the applicant in order for the staff to perform an independent radiological consequence analysis. (A) DCD Tier 2, Revision 1, Sections 15.4.9.2.2 and 15.4.9.5.4 list "Identification of Operator Actions" and "Assumptions to be Confirmed by the COL Applicants," respectively. Please state if any of these actions and assumptions are credited in the radiological consequence analysis, and if they are included in the ESBWR technical specifications, and/or COL Action Items.

RAI Number	Reviewer	Question Summary	Full Text
			(B) Please state the break flow rate and break flow duration used in the radiological consequence analysis. (C) Please provide a copy of dose calculation performed including determination of iodine appearance rates and resulting iodine concentrations due to the iodine spike. (D) Please provide control room operator doses for both with an assumed pre-accident iodine spike and an accident-initiated iodine spike.
15-4-5	Lee J	Additional information is required for the fuel building design and configuration to preclude a postulated spent fuel cask drop.	Additional information is required for the fuel building design and configuration to preclude a postulated spent fuel cask drop. (A) DCD Tier 2, Revision 1, Section 15.4.10.1 states that the fuel building design is such that a spent fuel cask drop height of 9.2 meter, as specified in SRP 15.7.5, is not exceeded. Please provide a copy of fuel building layout showing the height of spent fuel cask transfer path in the fuel building. (B) DCD Tier 2, Revision 1, Section 15.4.11 lists "COL information." Please state if any of items listed is a COL Action Item or an ITAAC item.

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