July 20, 2007

Mr. Robert E. Brown
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd MC A-45
Wilmington NC 28401

SUBJECT: ECONOMIC SIMPLIFIED BOILING WATER REACTOR (ESBWR) CHAPTER 15 OPEN ITEMS

Dear Mr. Brown:

As you are aware, the U.S. Nuclear Regulatory Commission staff is preparing the safety evaluation report (SER) for the ESBWR design certification application submitted by GE-Hitachi Nuclear Energy Americas LLC (GHNEA) on August 24, 2005. The staff has identified 52 open items for SER Chapter 15, "Transient and Accident Analysis," which are enclosed for your information. The staff is prepared to review your responses to the open items and have conference calls and meetings with your staff, as appropriate, to resolve these open items to support issuance of the SER.

Please provide a response date for any late or unscheduled open items discussed in the enclosure.

This open item letter is based on the staff's review of the ESBWR Design Control Document (DCD) Revision 3, Request for Additional Information (RAI) responses and other submittals received as of July 13, 2007. The staff will continue its review as additional RAI responses and other deliverables are submitted, including future DCD Revisions. The staff will inform cognizant GHNEA staff of any resulting changes to the status of Chapter 15. If you have any questions, please contact Amy Cubbage at (301) 415-2875 or aec@nrc.gov or Bruce Bavol at (301) 415-6715 or bmb2@nrc.gov.

Sincerely,

/RA/

Mohammed A. Shuaibi, Chief
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No.: 052-010

Enclosure:
As stated

cc: See next page

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cc: See next page
ADAMS ACCESSION NO. ML071840010

OFFICE	PM:NGE1	PM:NGE1	BC: SRSB	BC: SRSB
NAME	BBavol	ACubbage	JDonoghue	GCranston
DATE	7/20/07	07/20/07	07/12/07	07/11/07
OFFICE	BC: SPLB	BC: RSAC	BC: NGE1	
NAME	HHamzehee	CCox	MShuaibi	
DATE	07/12/07	07/11/07	07/20/07	

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General Electric-Hitachi ESBWR
Preliminary Open Item
Chapter 15
Transient and Accident Analysis

RAI 15.0-2, Supplement No. 1, 06/06/07, ML071580005

(A) GE's response to RAI 15.0-2 states "Control Rod Drive System (CRDS): The high pressure makeup water function of this system is credited in several event scenarios as backup level control to feedwater. This function of CRDS is nonsafety-related. If credit is not taken for the high pressure makeup water function of the CRDS, then the Isolation Condenser System and Gravity-Driven Cooling System would ensure acceptable inventory control." GE's response to RAI 16.2-33 stated that "Both the RAI 15.0-2 and the RAI 16.0-1 responses indicated that this function is not in the primary success path for mitigating transients and accidents because the safety-related isolation condenser and gravity-driven cooling system will ensure water inventory is maintained within the acceptance criteria for the applicable event even if the nonsafety related CRD system makeup water function failed." The DCD should be revised to include this information and to include the results of analysis that support this conclusion.

(B) Add a table in Section 15.0 of DCD Tier 2 listing the following non-safety related equipment that is credited in the AOO, infrequent event and/or accident analyses: Control Rod Drive System -Makeup Water Control Rod Drive System-SCRRI (included in the TS) Fuel and Auxiliary Pool Cooling System Feedwater Control System RC& IS (RWM and ATLM are included in the TS) Steam Bypass and Control System (included in the TS).

Status: GHNEA committed to provide a response by 11/28/07.

RAI 15.0-12, Supplement No. 1, 06/21/07, ML071720041

In Rev. 3 of DCD 15.0-3 it is stated "Where an acceptance criterion is not specified in regulations and the SRP, then the criterion in the Reference 1 LTR shall be used." The LTR NEDO-33175, Rev.1 has not been approved by the staff. Revise the DCD to reference only the regulations and the SRP. Any departures from the SRP should be justified in the DCD rather than referring to NEDO-33175.

Status: GHNEA committed to provide a response by 08/31/07.

RAI 15.0-15, 05/09/06, ML061220012

(A) Please describe the basis for the classification of the RWE including initiating actions/ events and mitigating strategies from all modes of operation.

(B) Please describe the potential for a "gang" withdrawal error (e.g. multiple control rods).

(C) Please identify the proposed acceptance criteria for the new event classification.

Status: GHNEA responded on 06/16/06, MFN 06-173.
GHNEA's response is under staff review, and will be discussed during an audit on July 30, 2007.

RAI 15.0-16, Supplement No. 1, 03/27/07, ML070860631

The staff reviewed the RAI 15.0-16 response and found the response unacceptable. It is our position that the SLMCPR numerical value should be kept as a safety limit in the TS as in the BWR STS. Our position is based on the following:

(1) Allowing the removal of the SLMCPR eliminates regulatory control of core analysis issues and eliminates a mechanism for the staff to apply conditions that might be needed in some situations to ensure safety. The NRC previously considered and rejected the same request (i.e., removal of the SLMCPR from the TS) from BWROG and Exelon (ML043140475 and ML030520480).

(2) Use of TRACG for calculating the OLMCPR is not an appropriate basis for removing the SLMCPR from the TS. In its response, GE referred to the ESBWR TRACG methodology used for the ESBWR OLMCPR calculation. GE states that this process allows for the direct calculation of the number of rods subject to boiling transition (NRSBT) for a transient. Since the SLMCPR is not used to calculate the OLMCPR, it is appropriate not to include the SLMCPR as assurance that the SAFDLs are met in TS. The staff disagrees. The staff does not find use of the TRACG methodology to calculate OLMCPR to be an appropriate basis for excluding the SLMCPR from the TS. The NRC has approved the TRACG methodology for calculating OLMCPR in the past for BWR/2-6s, and licensees who currently use the TRACG methodology for calculating OLMCPR are still required to have a SLMCPR TS.

(3) 10 CFR 50.36c (1)(i)A specifically states, “Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.” The staff has interpreted this section as requiring that the values of the safety limits must remain in a licensee’s technical specifications. Revised TS Section 2.1.1.2 (Rev. 3) proposes to replace the MCPR safety limit values with a description of what the safety limit protects against, i. e.; “Greater than 99.9% of the fuel rods in the core would be expected avoid boiling transition.” The proposed description is a fuel condition and is not an acceptance criterion. The staff does not believe that the proposed change is consistent with the staff’s interpretation of section 50.36c(1)(i)A since it is not a safety limit, but a criterion.

Status: GHNEA committed to provide a response by 09/21/07.

RAI 15.0-17, Supplement No. 1, 10/11/06, ML062830003

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.

Status: GHNEA responded on 06/15/07, MFN 06-418 Supplement 1. GHNEA's response is under staff review.

RAI 15.0-18, 10/11/06, ML062830229

DCD Tier 2 Appendix 15A provides an analysis to determine the frequency of occurrence of events classified as infrequent events. Section 15A.4 lists four "analysis assumptions" that are to be confirmed by the COL applicant:

- The feedwater control system (FWCS) is equipped with a triple-redundant, fault-tolerant digital controller (FTDC) including power supplies, and input/output signals. It is required that the Mean Time to Failure (MTTF) of the Feedwater System Controller be higher than 1000 years. Compliance to this requirement should be established through a reliability analysis by the vendor for the controller.

- The steam bypass and pressure control (SB&PC) system is equipped with a triple-redundant, fault-tolerant digital controller (FTDC) including power supplies, and input/output signals. It is required that the Mean Time to Failure (MTTF) of the SB&PC Controller be higher than 1000 years. Compliance to this requirement should be established through a reliability analysis by the vendor for the controller.

- The reactor water cleanup (RWCU)/shutdown cooling (SDC) system shall be designed with an interlock that prevents accidental engagement of the system in shutdown cooling mode when the reactor is in operation. The interlock feature shall be designed to be single-failure proof.

- No single failure in the nitrogen system can lead to an Inadvertent Opening of a Safety/Relief Valve.

  (a) It is anticipated that the COL applicant would not be able to submit a reliability analysis of this equipment since it would not yet be procured. Explain the rationale for requiring that the COL applicant confirm the reliability of equipment rather than performing bounding assessments of the event frequencies, based on high level design features and principles.

  (b) Since the event frequencies provided in Appendix 15A form the basis for the categorization of events as "infrequent events" rather than anticipated operational occurrences, these key design features and principles should be included in Tier 1 design descriptions and ITAAC should be provided to verify them.

  (c) Add a COL information item to include the FWCS and the SB&PC in the design reliability assurance program (D-RAP) to ensure that the COL holder will evaluate the reliability of these components and determine if the equipment is acceptable or if it must be redesigned to achieve a lower failure rate.

Status: GHNEA committed to provide a response by 08/30/07.

RAI 15.0-24, 10/11/06, ML062830229

The analysis performed to estimate the frequency of event “Inadvertent Opening of a Depressurization Valve” relies heavily on PRA modeling of the I&C systems. This includes assumptions about hardware and software CCFs. The staff is waiting for the applicant’s response to Chapter 19.1 (PRA) RAIs (#19.1-21 and #19.1-22) to continue the review of DCD Tier 2, Revision1, Section 15A.3.9.

A brief discussion in Subsection DCD Tier 2, Revision1, 15A.3.9.2.2.7, on the frequency contribution of CCFs, does not provide the information the staff needs to understand how I&C hardware and software CCFs were modeled. Please confirm that the responses to RAIs 19.1-21 and 19.1-22 will address questions related to PRA modeling of the I&C system, including CCFs, which also will clarify the analysis in DCD Tier 2, Revision1, Section 15A.3.9. If not, please provide additional information in DCD Tier 2, Appendix 15A to address this concern.

Status: GHNEA responded on 05/16/07, MFN 07-289.
GHNEA’s response is under staff review.

RAI 15.0-26, 05/30/07, ML071490166

In DCD Tier 2, Rev 3, Section 15.0.1.2 (3), “ An infrequent event is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence of < 1/100 per year, and a radiological consequence less than an accident.”

According to regulations, there are only two event categories: anticipated operational occurrences (AOOs) and Accidents. The staff believes that the infrequent event (IE) category is a subset of the accident category, and hence the radiological consequence is less than of a design basis accident. Hence revise the following from “ -----radiological consequence less than an accident” to “ —radiological consequence less than a design basis accident .”

Status: GHNEA committed to provide a response by 08/31/07.

RAI 15.0-27, 05/30/07, ML071490166

GE transient analyses used evaluation model, TRACG, NEDC-33083P-A, “TRACG Application for ESBWR” to analyze most of the AOOs and IEs in Chapter 15. In reference documents, demonstration of calculations for ESBWR AOOs are presented. However, there is no discussion of Infrequent Events analyses in this LTR. Is there a plan by GE to submit additional information to qualify TRACG for analyzing the Infrequent Events?

Status: GHNEA committed to provide a response by 11/30/07

RAI 15.0-28, 06/21/07, ML071590313

In Section 15A.3.10, “Stuck Open Relief Valve”, GE estimates this initiating event frequency by taking credit for the availability of the Isolation Condenser (IC) System for the ESBWR. It is assumed that the probability of the IC being unavailable is less than 0.1. There is no justification for this number in this section. Please provide the technical basis for this number.

Status: GHNEA committed to provide a response by 09/10/07.

RAI 15.0-29, 06/21/07, ML071590313

In Section 15A.3.10, GE provides a best estimate value for the expected frequency of a stuck open SRV in an ESBWR of 3.28E-04/yr. However, the traditional number used for existing BWR plants is about 4.6E-2/yr (NUREG/CR-5750). In addition, the number used in the ESBWR PRA is 2.23E-2/yr (NEDO-33201 Rev.2, Section 2). Please explain why the best estimate ESBWR frequency (i.e., 3.28E-04/yr) was not used in the ESBWR PRA.

Status: GHNEA committed to provide a response by 09/10/07.

RAI 15.2-2, Supplement No. 1, 07/02/07, ML071830081

DCD Figures 15.2-4(a) and 15.2-7(a) show a high narrow power peak of less than a second duration. Energy deposition has not been calculated to assure of acceptable fuel cladding interaction. Please explain why you did not consider fuel energy deposition.

GE’s response to this RAI stated that “this event scenario will be studied in more detail and revisions will be made to the DCD as appropriate.” Please provide a supplemental response to document the completion of this study and inform the staff if any changes result.

Status: GHNEA has not committed to a response date.

RAI 15.2-5, 10/11/06, ML062830003

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.

Status: GHNEA committed to provide a response by 12/07/07.

RAI 15.3-4, Supplement No. 1, & 15.3-5, Supplement No. 1, 05/21/07, ML071410293

These RAIs are being tracked under RAI 2.4-29 S01.

The applicant needs to revise DCD Section 15.3.16 in response to RAIs 15.3-4 and 15.3-5 to clarify that sealing of concrete walls cannot be relied upon to contain the release of all of the liquid radwaste in the compartment, since SRP Section 11.2 and BTP 11-6 preclude this. According to BTP 11-6, the applicant needs to provide additional details on "special design features" to support their statement and update the radiological assessment accordingly. These are features beyond those discussed in RG 1.143. As is stated in DCD Rev. 3, Section 15.3.16, the approach relies on the use of "a sealant" or "sealed concrete walls" as the mitigating feature of the design. However, BTP-11- 6 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 the building foundation." Accordingly, the applicant is requested to update the basis and assumptions used in the analysis presented in DCD Tier 2, Rev. 3, Section 15.3.16 to be consistent with the SRP guidance in BTP 11-6; and discuss why the release of the postulated inventory of radioactive materials to surface or ground water is not limiting in the analysis as compared to the current case where the volatile airborne fraction of radioactivity (as radioiodines) is assumed to be released in the environment; and update the text and tables in DCD Rev. 3, Sections 15.3.16 and 2.4.13, and Table 2.0-2 of DCD Rev. 3, Section 2.0.

Status: GHNEA committed to provide a response by 07/21/07.

<u>RAI 15.3-11, Supplement No. 1, 07/02/07, ML071830081</u>

(1) DCD Figure 15.3-4a indicates a sharp rise in total power (although the corresponding simulated power peak is not as pronounced) very much like control rod drop event. Please calculate the total power deposition and the corresponding cladding strain along with the pellet clad mechanical interaction for cladding strain.

(2) Generator load Rejection with Total Turbine Bypass Failure event results in a very short burst of energy. Please calculate the fuel energy deposition and the pellet cladding mechanical interaction for this transient.

Status: GHNEA has not committed to a response date.

<u>RAI 15.3-14, 10/11/06, ML062830003</u>

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?

Status: GHNEA responded on 05/22/07, MFN 07-264.
GHNEA’s response is under staff review.

<u>RAI 15.3-16, 10/11/06, ML062830003</u>

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.

Status: *GHNEA responded on 05/22/07, MFN 07-264.*
GHNEA's response is under staff review.

RAI 15.3-19, 10/11/06, ML062830003

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.

Status: *GHNEA committed to provide a response by 01/11/08.*

RAI 15.3-23, Supplement No. 1, 07/09/07, ML071970171

Note that the following references of the DCD were used in the follow-up to the response of RAI 15.3-23.

a) Subsection 15A.3.8.2, regarding probability analysis of safety relief valve malfunction,

b) Subsection 15A.3.10, regarding the probability of a stuck open relief valve,

c) Subsection 15.3.15, regarding stuck open relief valve event summary description,

d) Subsection 3.9.1.4, regarding evaluation of faulted conditions,

e) Enclosure from MFN 07-011,

RAI 15.3-23 covered most of the questions raised in the RAI but concluded that no DCD changes are necessary. Please consider the following two instances and <u>modify the DCD to provide the justification of the following comments</u>.:

1 Reference (d) above, regarding ASME Class 2 and 3 Valves states: "Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions.... ...are obtained from NC/ND-3400 the Code. These allowables are above elastic limits."

Reference (e) above states: "The acceptance criteria.... require... that after testing ....the valve shall not exhibit any deformation that would degrade its performance beyond the specification prescribed limits." The main cause of valve malfunction after opening is failure to reseat properly due to deformation. If the allowable criteria are beyond the elastic limit how do you expect the valve to reseat properly?

Please clarify these statements.

2 Reference (a) above, discussing operator error states: "He should not be opening the SRVs inadvertently and he cannot do it accidentally because a deliberate action is required to open the SRVs." This statement does not make sense, please revise the DCD to clarify this statement and to provide justification to support the conclusion that the probability of an inadvertent opening of a relief valve resulting from operator action is judged to be negligible.

Status: GHNEA has not committed to a response date.

RAI 15.3-25, Supplement No. 1, 06/13/07, ML071580274

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006
GE response in MFN 07-017 dated February 16, 2007

(1) Concerning GE's response to Item B:

a) Add English Unit (curies) in the table or provide a new separate table with English units only.

b) Reference DCD Tier 2, Revision 3, Appendix 15B, "LOCA Inventory," as responded to RAI 15.4-9.

c) Revise to read fuel exposure as "35 GWd/MTU core average."

(2) Revise Table 15.3-13 as follows:

a) Add number of fuel rods in core, condenser leak rate, release duration, and release points, to the table.

b) Correct typographical error to read Table 15.3-16 (instead of Table 15.4-19).

c) Add justification for the use of radial peaking factor of 1.5 for the 1000 fuel rods failed. What is the peak fuel rod average burn up? Is this radial peaking factor specified in ESBWR Technical Specifications?

d) Show or reference the control room χ/Q values provided in Table 15.3-13 as"1000 fuel rod failure parameters" in DCD, Tier 2, Chapter 2, Table 2.0-1.

e) Add justification for the amount of iodine, noble gases, and alkali metals released from the failed fuel rods. Does it meet the maximum linear heat generation rate specified in Footnote 11 of Regulatory Guide 1.183, Table 3? (See DCD, Tier 2, Revision 3, Tables 6.3-1 and 6.3-11 for bounding peak linear heat generation rate specified).

(3) Revise Table 15.3-16 as follows:

a) Add the control room operator doses.

b) Reword EAB to read "... for any (worst) 2 hours" rather than "for the entire period of the radioactive cloud passage."

Status: *GHNEA committed to provide a response by 09/21/07.*

RAI 15.3-26, 05/30/07, ML071490166

Since only the limiting events will be analyzed during the COL licensing phase, analyses of all Infrequent Events are required for design certification. DCD Tier 2 Table 15.3-1 needs to be revised to show the results of all the Infrequent events. Events described in Sections 15.3.7 to 12 and 15.3.14 needs to be analyzed.

Status: GHNEA committed to provide a response by 10/19/07.

RAI 15.3-27, 05/30/07, ML071490166

In DCD Tier 2, Rev 3, Section 15.2.6, GE states, “For the core loading in figure 4.3-1, the resulting initial core MCPR operating limit is 1.30.” The staff is unclear as to whether this is OLMCPR or SLMCPR. Please provide the basis for the value (1.30).

Status: GHNEA committed to provide a response by 10/19/07.

RAI 15.3-28, 05/30/07, ML071490166

Add “ Minimum Critical Power Ratio” to the following DCD Figures:
Figure 15.3-1g, 15.3.2g, 15.3.3g, 15.3.4g, 15.3.5g, 15.3.8g and 15.3.9g

Status: GHNEA committed to provide a response by 11/02/07.

RAI 15.3-29, 05/30/07, ML071490166

In DCD Tier 2, Rev 3, Section 15.3.3.1, Pressure Regulator Failure-Opening of All Turbine Control and Bypass Valves, it is stated that “-----the event is considered as a limiting fault.” The staff does not agree with this characterization of the event. This event is an Infrequent category event as referred in other parts of the DCD and should be characterized as an Infrequent Event not as limiting fault. Revision of the DCD is required.

Status: GHNEA committed to provide a response by 10/19/07.

RAI 15.3-30, 05/30/07, ML071490166

For the Control Rod Withdrawal Error During Start-up event the description of the method in DCD 15.3.8.3.1 does not state whether the adiabatic heating assumption is conservative as far as reactivity is concerned. If the heat is restricted in the fuel, it will exaggerate the Doppler feedback and eliminate the density feedback. In reality, both feedbacks will be present. Explain why the adiabatic assumption produces conservative results?

Status: GHNEA committed to provide a response by 09/21/07.

RAI 15.3-31, 05/30/07, ML071490166

The calculation of the event frequency (Section 15A.3.8) assumes 0.0 /pry frequency for the following: incorrect set point or spring adjustment; spring relaxation; and operator error. Is the operator error referring to an error in setting or adjusting the valve spring or some other operator error? Please justify the above values in light of operating experience associated with SRVs of a similar design to ESBWR.

Status: GHNEA committed to provide a response by 09/10/07.

<u>RAI 15.3-32, 05/30/07, ML071490166</u>

DCD Tier 2, Rev 3, Section 15.2.0 lists several “COL applicant assumptions” that are applied in the TRACG calculations. A COL information item was provided in DCD Section 15.2.7 for the COL applicant to confirm the applicability of these assumptions. Since the assumptions are also applied to DCD sections 15.3 and 15.5.5, similar COL information items should be added to DCD sections 15.3.17 and 15.5.8 for completeness.

Status: GHNEA committed to provide a response by 08/17/07.

<u>RAI 15.3-33, 05/30/07, ML071490166</u>

Control Rod Withdrawal Error During Power Operation, DCD 15.3.9.4 states: “An evaluation of the barrier performance is not made for this event, because there is no postulated set of circumstances for which this event could occur.”

The current version of the SRP requires that RWE be analyzed as an AOO. The additional system in the ESBWR is supposed to prevent RWE but the estimated frequency is considerably smaller than 1.0E-02, it is finite and accounts for ATLM failure. Other transients, an example being 15.3.6 with an estimated frequency in the same order of magnitude as this one were analyzed, and therefore the staff believes that the RWE transient should be analyzed. Provide your analysis of this event.

Status: GHNEA committed to provide a response by 10/19/07

<u>RAI 15.3-34, 07/18/2007, ML071990395</u>

Regarding the inadvertent shutdown cooling function operation event, DCD Tier 2, Rev. 3, Section 15.3.12 states that “the increased subcooling caused by mis-operation of the RWCU/SDC shutdown cooling mode could result in a slow power increase due to the reactivity insertion. During power operation the reactor settles in a new steady state. During startup or shutdown, this power rise is terminated by a flux scram before fuel thermal limits are approached. Therefore, only a qualitative description is provided here and this event does not have to be analyzed for a specific core configuration.”

Please quantify the range of expected temperature limits and the resulting reactivity and reactivity-rate resulting from mis-operation of the RWCU/SDC shutdown cooling mode to justify the statement that the <u>thermal limits will not be violated</u>.

Status: GHNEA has not committed to a response date.

<u>RAI 15.4-1, Supplement No. 1, 06/12/07, ML071580274</u>

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006
GE response in MFN 07-017 dated February 16, 2007

(1) In DCD, Tier 2, Revision 3, Section 15.4.1, GE stated that two scenarios of the fuel handling accident were postulated: drop of a raised fuel assembly (1) onto the reactor core and (2) into the spent fuel storage pool. Provide the radiological consequence analysis for

each scenario complete with fission product release pathways to the environment. State which scenario is bounding and why. Include this information in the DCD.

(2) Please state if containment, reactor building, and/or fuel building are required to maintain its integrity during fuel handling operation. Do you consider this requirement as a COL action item? State in DCD how you satisfy the guidance provided in Footnote 2 of Appendix B in Regulatory Guide 1.183.

(3) In DCD, Tier 2, Table 2.0-1 and Table 15.4-2, provide the EAB, LPZ, and control room χ/Q values used for each release point.

(4) State in the DCD that the control room is not isolated during this event and that the normal control room ventilation system will be in operation.

(5) State in DCD the amount of iodine, noble gases, and alkali metals released from the failed fuel rods. Does it meet the maximum linear heat generation rate specified in Footnote 11 of Regulatory Guide 1.183, Table 3? (See DCD, Tier 2, Revision 3, Tables 6.3-1 and 6.3-11 for bounding peak linear heat generation rate specified).

(6) Justify in the DCD the use of radial peaking factor of 1.5 for the 1000 fuel rods failed. What is the peak fuel rod average burn up?

(7) Response to Item A of RAI 15.4-1

a) Reconstruct the table showing fission product inventory in curies and reference to DCD, Tier 2, Appendix 15B.

b) State the total number of fuel bundles in the core and DF of 200 used as notes to the table.

c) Correct typographical error to read RPF (not RFP) in note.

(8) Response to Item C of RAI 15.4-1

a) State which sets of the control room χ/Q values in the table were used for this event.

b) Add the “Fuel Building Cask Door to Control Room Air Intake” to the DCD, Tier 2, Tables 2.0-1 and Table 15.4-2, if used for this event.

(9) Response to Item E of RAI 15.4-1

a) State in the DCD which release pathway is bounding and why.

(10) Response to Item J of RAI 15.4-1

a) State in the DCD where and how the control room χ/Q value of 1.0E-3 s/m_3 were used for this event.

b) GE stated that the control room normal air intake flow rate and the control room habitability area volume are ITAAC items. Reference sections and ITAAC table numbers in DCD tier 1.

(11) Revise Table 15.4-4

a) Delete "Within Containment" from the table (a typographical error)

b) Recalculate LPZ doses using LPZ χ/Q values from 0 to 30 days.

c) The LPZ dose should be "for 0 to 30 days."

Status: *GHNEA committed to provide a response by 08/21/07.*

RAI 15.4-2, Supplement No. 1, 06/12/07, ML071580274

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006
GE response in MFN 07-100 dated March 26, 2007

(1) Provide revised steam and water mass releases for the main steam line break accident.

(2) Add the following information to Table 15.4-11:

Duration of accident
EAB, LPZ, and control room χ/Q values
Release point
Control room operator doses
Control room not isolated
Control room normal ventilation system will be in operation during this event

(3) Revise the following information in Table 15.4-13:

a) Reword EAB to read "... for any (worst) 2 hours" rather than "for the entire period of the radioactive cloud passage."

b) The LPZ dose should be "for 0 to 30 days."

c) Provide control room operator doses for pre and post-iodine spike.

Status: *GHNEA committed to provide a response by 09/05/07.*

RAI 15.4-3, Supplement No. 1, 06/12/07, ML071580274

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006
GE response in MFN 07-100 dated March 26, 2007

(1) Add the following information to Tables15.4-14 and 15.4-17:

Duration of accident
EAB, LPZ, and control room χ/Q values
Release point
Control room operator doses
Control room not isolated
Control room normal ventilation system will be in operation during this event

(2) Revise the following information in Tables15.4-19 and 23:

a) Reword EAB to read "... for any (worst) 2 hours" rather than "for the entire period of the radioactive cloud passage."

b) The LPZ dose should be "for 0 to 30 days."

Status: *GHNEA committed to provide a response by 09/05/07.*

<u>RAI 15.4-4, Supplement No. 1, 06/12/07, ML071580274</u>

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006
GE response in MFN 07-197 dated April 12, 2007

(1) Add the following information into Table 15.4-21

Duration of accident
EAB, LPZ, and control room χ/Q values
Release point
Control room operator doses
Control room not isolated
Control room normal ventilation system will be in operation during this event

(2) The revisions of Table 15.4-19 and 23 as indicated in Supplemental No. 1 RAI 15.4-3 above are also required for resolution of this RAI.

Status: *GHNEA committed to provide a response by 10/06/07.*

<u>RAI 15.4-6, 01/29/07, ML070230300</u>

Please incorporate the radiological consequence analyses provided in (1) General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006," (LTR) and (2) General Electric Research Report No. VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment - Part 1, October 2006," (VTT report) into DCD Section 15.4.4, "Loss-of-Coolant Accident Inside Containment Radiological Analysis," or please incorporate these two reports into the DCD Chapter 15 as appendices.

Status: GHNEA committed to provide a response by 09/20/07.

RAI 15.4-7, 01/29/07, ML070230300

The General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006," (LTR) includes the radiological consequence analyses, complete with fission product removal rates in the containment for Accident Scenario 1, "Reactor Bottom Drain Line Break—With automatic depressurization system (ADS) (Low Pressure) Scenario."

Complete and provide the same radiological consequence analyses, complete with fission product removal rates in the containment and associated pH evaluation for Accident Scenario 2, "Reactor Bottom Drain Line Break—without ADS (High Pressure) Scenario," and Accident Scenario 3, "Loss of Preferred Power," as discussed in Section 1.3, "Accident Scenarios Evaluated" of the LTR. Please incorporate these two remaining radiological consequence analyses into the LTR and Section 15.4.4. Compare and discuss the results of the radiological consequences and fission product removal rates in the containment for Accident Scenarios 1, 2, and 3.

Status: GHNEA committed to provide a response by 09/20/07.

RAI 15.4-8, 01/29/07, ML070230300

Proposed DCD Tier 2, Revision 3, Section 15.4.4 states that its radiological consequence analyses are based on NUREG-1465 alternative source terms and the methodology in Regulatory Guide (RG) 1.183. However, Section 15.4.4.2.1 states that the core remains covered throughout the accident and there is no fuel damage. This is inconsistent with NUREG-1465 and RG 1.183.

Please rectify the inconsistencies in these statements. Review the entire General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR) and Section 15.4.4 to ensure that there are no further discrepancies.

Status: GHNEA committed to provide a response by 09/21/07.

RAI 15.4-13, 01/29/07, ML070230300

Proposed DCD, Tier 2, Revision 3, Section 15.4.4.5.2.2 (last paragraph) and Section 4.1.2.1, "Cesium Hydroxide," (CsOH) of the General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR) discusses the production and formation of CsOH stating that "The cesium that is not in the chemical form of CsI is assumed to exist .... in the form of CsOH." The staff believes cesium may also exist in the form of cesium compounds other than CsOH (i.e., cesium molybdate, cesium manganate). Cesium may enter containment in the form of CsOH, cesium borate or cesium iodide. Although CsOH is highly soluble in water and a strong base, by itself it is not sufficient to maintain pH in the containment pools above 7. Given your statement that pH in the containment pool will remain alkaline due to sufficient amount of CsOH, provide a sensitivity analysis of pH to CsOH formation (zero to 100 percent formation).

Status: GHNEA committed to provide a response by 10/15/07.

RAI 15.4-15, 01/29/07, ML070230300

Section 4.1.2.1, “Hydrochloric Acid ,”(HCl) in the General Electric Licensing Topical Report, NEDE-33279, “ESBWR Containment Fission Product Removal Evaluation Model, October 2006,”(LTR) discusses the production and formation of Hcl. The formation of organic iodides in the containment following a loss-of-coolant accident could result from processes initiated at the surfaces of containment paint. Therefore, please provide paint specification for paint to be used in the containment surfaces.

Status: GHNEA committed to provide a response by 10/15/07

RAI 15.4-16, 01/29/07, ML070230300

(A) In the "Abstract" and Section 1.1, "Background," of the General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR), revise the current statement." The passive systems are different from those used in current generation BWRs, thus many of the regulations and methodologies used in previous analyses are not directly applicable to the ESBWR design" with "The passive systems are different from those used in current generation BWRs, thus certain regulations (source terms) and methodologies used in previous analyses are not directly applicable to the ESBWR design."

(B) In Section 1.1, "Background" of the LTR, delete the following sentence, "This research often led...... based on TID source term."

(C) In Section 1.3, "Accident Scenarios Evaluated," of the LTR, revise the word from "credited" to "provided" to read "No active systems, such as safety related containment sprays or Standby Gas Treatment Systems are provided to limit...."Number -5-Enclosure.

Status: GHNEA responded on 07/10/07, MFN 07-302.
GHNEA’s response is under staff review.

RAI 15.4-17, 01/29/07, ML070230300

Section 1.3, “Accident Scenarios Evaluated,” of the General Electric Licensing Topical Report, NEDE-33279, “ESBWR Containment Fission Product Removal Evaluation Model, October 2006,”(LTR), describes three accident scenarios evaluated for the radiological consequence analysis. Describe each of the three accident scenarios in more detail, complete with sequence of events, operation and availability of the engineered safety features systems including the suppression pool, fission product transport pathways, and fission product release timing.

Status: GHNEA committed to provide a response by 09/21/07.

RAI 15.4-18, Supplement No. 1, 06/13/07, ML071580274

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006
GE response in MFN 07-198 dated April 12, 2007

(1) Provide revised or new accident model diagram shown in Enclosure 2 of the response, complete with all fission product transport and release pathways.

Status: GHNEA committed to provide a response by 10/19/07.

RAI 15.4-20, 01/29/07, ML070230300

The General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR), Appendix A, "LOCA Dose Assumptions," note 2, states that main steam line and main steam drain line data are not provided because fission product deposition in these lines are not credited. To perform an independent confirmatory radiological consequence analysis on this release pathway, the staff needs following information:

(1) main steam pipe diameter, length (horizontal), and volume between MSIVs, drain line valves, and drain line header to the condensers,

(2) outside diameter and thickness for the main steam lines, main steam drain lines, steam drain header to the condensers,

(3) insulation thickness, and

(4) steam pressure and temperature in the main steam lines, steam drain lines, steam drain header to the condensers.

Status: GHNEA committed to provide a response by 07/31/07.

RAI 15.4-22, 01/29/07, ML070230300

Proposed DCD, Tier 2, Revision 3, Table 15.4-5, "Loss-of-Coolant Accident Parameters," shows the fraction of condenser volume involved in iodine removal as 20 percent and the iodine removal factors as 99.5 percent. Please provide the model, methods, and assumptions used for fission product removal in the main condensers and justify the use of TID-14844 source term for this pathway for estimating its radiological consequences. Provide an estimated condenser vertical and horizontal surface areas available for aerosol deposition.

Status: GHNEA committed to provide a response by 10/15/07.

RAI 15.4-24, 01/29/07, ML070230300

Section 4.4, "Main Steam Isolation Valve Leakage," of the General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR) and Section 5.0, "Offsite Dose Calculations" stated that "fuel damage does not occur until ...." This is contrary to the guidance provided in RG 1.183 alternative source term (AST). Explain how fission product release timing for the radiological consequence analyses was determined and used throughout the LTR and Section 15.4.4 of the proposed DCD, Tier 2, Revision 3.

Status: GHNEA committed to provide a response by 09/21/07.

RAI 15.4-25, 01/29/07, ML070230300

Section 4.5, "Containment and Reactor Building Leakage Paths," of the General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR) states that "any leakage [steam/air/water mixture] from the [passive containment cooling] PCC condensers will be included in the overall containment leakage term (for the radiological consequence analyses)." The next sentence states that "liquid leakage from the PCC condensers and associated piping is not considered....."

Please explain the discrepancies. The staff believes any volatile fission products in the liquid leakage will leak from the reactor building to the environment.

Status: GHNEA responded on 07/10/07, MFN 07-302.
GHNEA's response is under staff review.

RAI 15.4-26, 01/29/07, ML070230300

Section 4.5, "Containment and Reactor Building Leakage Paths," of the General Electric Licensing Topical Report, NEDE-33279, "ESBWR Containment Fission Product Removal Evaluation Model, October 2006,"(LTR) stated that your radiological consequence analysis assumed an overall reactor building leakage rate of 50 percent per day.

(A) Provide the flow paths to be isolated and the method to be used to verify the leak rate.

(B) State whether the leakage rate test to meet the 50 percent per day limit is specified in the ESBWR Technical Specification (TS).

(C) Include this leak rate verification in Tier 1 as an ITAAC item to be confirmed at the COL stage. Section 6.2.3.1 "Design Bases," stated that "The RB is capable of periodic testing to assure that the leakage rates assumed in the radiological analyses are met."

Status: GHNEA responded on 06/27/07, MFN 07-299.
GHNEA's response is under staff review.

RAI 15.4-28, 01/29/07, ML070230300

In order to complete its evaluation, the staff needs to review the general assumptions and calculations used to prove that the containment sump pH will be maintained above 7 for 30 days following a loss-of-coolant accident (LOCA).

(A) Please provide this information by completing the attached table (see below) for each pool in the ESBWR design (Reactor Pressure Vessel, Lower Dry Well, Gravity Driven Cooling System, and Wet Well) in sufficient detail for the staff to perform independent calculations.

(B) In addition to completing the table for each pool, please complete the attached table for each different pH cases A through F as presented in General Electric Research Report No. VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment - Part 1, October 2006."

(C) In addition, please discuss the injection time of the buffer solution (sodium pentaborate) and its distribution from the moment it is injected to the time of 30 days after the accident.

Status: GHNEA committed to provide a response by 09/20/07.

RAI 15.4-29, 01/29/07, ML070230300

General Electric Research Report No. VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment - Part 1, October 2006," (VTT report) Section 3,"Goal," stated that "The purpose of this work is to investigate the capacity of [passive containment cooling system] PCCS condenser to remove airborne fission products from the containment atmosphere." The report concluded in Section 1, "Executive Summary," stating that "Both the experimental and modeling result suggest that [aerosol] deposition by diffusiophoresis could remove as much as 50 percent of particles from the gas flow. However, aerosol did not accumulate to the heat exchanger. In all experiments condensed water rinsed deposited particles from the tube walls.......". Therefore, GE's analysis implicitly assumed that the PCCS drove the iodine in aerosol form into the condensate and that it never came out of the solution with basic pH.

Explain the iodine transport phenomena in the containment and perform a rate analysis of steady state iodine transport within the containment including iodine re-volatilization (iodine production) from the melted fuel in the intact reactor pressure vessel and iodine removal by the PCCS condensers (iodine sink).

Status: GHNEA committed to provide a response by 09/20/07.

RAI 15.4-30, 05/10/07, ML0712303890

In response to RAI 15.4-30 - DCD Tier 2, Revision 3, Table 15.4-5, "Loss-of-Coolant Accident Parameters," specifies an assumed control room unfiltered air inleakage rate of 1.13E-2 cubic meter per minute. Please include this assumed control room unfiltered air inleakage rate (1) in DCD Tier 1, Table 2.16.2-1, "ITAAC for the Reactor Building HVAC," of Section 2.16.2, "Heating, Ventilating, and Air-Conditioning Systems," as an ITAAC item, and (2) in DCD Tier 2, Chapter 16, "Technical Specifications," Section 3.7.2, "Control room Habitability Area Heating, Ventilation, and Air Conditioning Subsystem," as surveillance requirements in accordance with guidance provided in Technical Specification Task Force (TSTF) - 448 (dated July 1, 2003).

Status: GHNEA committed to provide a response by 09/28/07.

RAI 15.4-31, 05/10/07, ML0712303890

In response to RAI 15.4-31 - In Revision 3 to the DCD, GE revised the control room χ/Q values in DCD Tier 1 Table 5.1-1 and Tier 2 Table 2.0-1, listing them as standard plant site design parameters. Two sets of control room χ/Q values are provided for reactor building, passive containment cooling system/reactor building roof, and turbine building release pathways; one set for unfiltered inleakage and the second set for the filtered air intake. Please state which set of control room χ/Q values are used for the control room radiological consequences and why.

Status: GHNEA committed to provide a response by 09/20/07.

<u>RAI 15.5-8, 10/10/06, ML062790238</u>

Regulatory Position 3.2.7 of RG 1.155 states that the ability to maintain appropriate containment integrity during a loss of all ac power should be addressed. The applicant addresses containment integrity in terms of design limits on pressures and temperatures. Please add a discussion to section 15.5 of the ESBWR DCD explaining what provisions are present to assure valve position indication and closure for containment isolation valves that may be in the open position at the onset of a station blackout.

Status: GHNEA committed to provide a response by 08/30/07.

cc:

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