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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 90 – Safety Analysis – RAI Numbers 15.4-9 through 15.4-12 and 15.4-14**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

D068

Reference:

1. MFN 07-084, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 69 Related to the ESBWR Design Certification Application*, January 30, 2007

Enclosures:

1. MFN 07-199 – Response to Portion of NRC Request for Additional Information Letter No. 90 – Safety Analysis – RAI Numbers 15.4-9 through 15.4-12 and 15.4-14

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	RE Brown	GE/Wilmington (with enclosures)
	eDRF	0065-5672R1

Enclosure 1

MFN 07-199

Response to Portion of NRC Request for Additional Information Letter No. 90 Related to ESBWR Design Certification Application

Safety Analysis

RAI Numbers 15.4-9 through 15.4-12, 15.4-14

NRC RAI 15.4-9:

Proposed DCD, Tier 2, Revision 3, Section 15.4.4.5.1.2 describes the reactor core inventory. Table 15B -1 provides reactor core fission product inventory. Please state the methods (i.e., computer code) used to obtain reactor core fission product inventory, complete with core thermal power, fuel burnup, and fuel enrichment. State whether the methods used are consistent with the guidance provided in Regulatory Guide (RG) 1.183, Section 3.1.

GE Response:

Appendix 15B of DCD, Tier 2, Revision 3 has been revised to include compliance information with Regulatory Guide 1.183, Section 3.1 for core thermal power, fuel burn-up, and fuel enrichment.

DCD Impact:

DCD Tier 2, Revision 4, Appendix 15B will include compliance information with Regulatory Guide 1.183, Section 3.1 for core thermal power, fuel burn-up, and fuel enrichment as shown on the attached markup.

<u>NRC RAI 15.4-10</u>:

Concerning Proposed DCD, Tier 2, Revision 3, Section 15.4.4.5.2,"Radionuclide Releases and Pathways," state whether the main steam isolation valve (MSIV) leakage to the turbine building is included in the total containment leakage rate of 0.5 percent per day by volume.

<u>GE Response</u>:

The MSIV leakage assumed in the LOCA dose consequence calculation is not included in the determination of the overall containment leakage rate (L_a). GE previously took an exemption to 10 CFR 50, Appendix J requirements for MSIV leakage testing program in response to RAI 6.2-75 (GE Letter MFN 06-231 dated July 18. 2006).

DCD Tier 2, Revision 3, Subsection 6.2.6.3 states "The measured leakage rate of MSIV in a Type C test will be excluded when determining the combined leakage rate of components subject to Type B and Type C tests. The justification for this exemption from 10 CFR 50 Appendix J requirement is because it is excluded from L_a which is redefined in Subsection 6.2.6.1.1."

DCD Tier 2, Revision 3, Subsection 6.2.6.1.1 states: "A standard statistical analysis of the data is conducted by a linear regression analysis using the method of least squares to determine the leakage rate and associated 95% Upper Confidence Limit (UCL). ILRT results are satisfactory if the UCL is less than 75% of the maximum allowable leakage rate, L_a. As an exemption from the definition of L_a in 10 CFR 50 Appendix J, the maximum allowable leakage rate (L_a) is redefined as Containment Leakage Rate given in Table 6.2-1 which excludes the MSIV leakage rate. The treatment of MSIV leakage pathway separately in radiological dose analysis in Section 15.4.4.5.2 justifies this exemption."

DCD Tier 2, Subsection 15.4.4.5.2 will be revised to clarify that MSIV leakage is not included in the overall containment leakage determination.

<u>DCD Impact</u>:

DCD Tier 2, Subsection 15.4.4.5.2, Revision 4 will include a clarification on MSIB leakage as noted on the attached markup.

NRC RAI 15.4-11:

Proposed DCD, Tier 2, Revision 3, Section 15.4.4.5.2, "Radionuclide Releases and Pathways, "states that "the remaining 2% of primary containment leakage [out of 0.5 percent] is assumed to leak through the [passive containment cooling system] PCCS into the airspace directly above the PCCS and [isolation condenser] IC pools. This leakage is quickly vented directly to the atmosphere [bypassing the reactor building enclosure]." The PCCS condensers are an extension of the containment boundary.

DCD, Tier 2, Revision 2, Section 6.2.2.2.2 states that "Spectacle flanges are included in the [PCCS condenser] drain line and in the [PCCS condenser] vent line to conduct post-maintenance leakage tests separately from Type A containment leakage tests. (See DCD Figure 6.2-16 in page 6.2-169) Include the leakage test for meeting the PCCS leakage limit (2 percent of La) in the ESBWR Technical Specification (TS), and in Tier 1, Table 2.15.4 -1, "ITAAC For The Passive Containment Cooling System."

GE Response:

The ESBWR Technical Specifications (TS), as documented in DCD, Tier 2, Revision 3, Chapter 16 addresses PCCS leakage assumptions per GE Topical Report NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model". The following DCD Tier 2, Chapter 16, TS sections apply to containment leakage tests:

- **TS 5.5.9.c** states: "The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.5% of containment air weight per day."
- **TS 5.5.9.d.1** states "Containment leakage rate acceptance criterion is {$\leq 0.98\ L_a$ for leakage from Containment into the Reactor Building and $\leq 0.02\ L_a$ for leakage through the Passive Containment Cooling System (PCCS)}."

The PCCS leakage value is also supported in the Bases for TS 3.6.1.1 and TS 3.6.3.1. TS SR 3.6.1.1.1 requires the licensee to "Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with Containment Leakage Rate Testing Program." SR 3.6.1.1.1 provides assurance that containment leakage (consisting of both leakage through the containment and via the PCCS) is less than or equal to 0.5% of containment air weight per day, with each limited to 0.98% and 0.02%, respectively. The Revision 3 ESBWR TS changes address the Appendix J exemption. No additional TS changes are warranted at this time.

Item 3 of DCD, Tier 1, Table 2.15.4-1, "ITAAC for the Passive Containment Cooling System," is being revised to reference the PCCS specific testing limit delineated in TS 5.5.9.d.1.

DCD Impact:

DCD Tier 1, Section 2.15.4 will be revised as noted on the attached markup.

NRC RAI 15.4-12:

Proposed DCD, Tier 2, Revision 3, Section 15.4.4.5.2.1 describes removal of elemental iodine from containment. Provide all numerical values used for each parameter in your equation for estimating natural deposition of elemental iodine in the containment.

GE Response:

The elemental iodine coefficient used in the LOCA dose consequence analysis documented in NEDE-33279P was based on guidance found in SRP 6.5.2. Specifically, the iodine removal rate constant for a particular compartment "n" was calculated based on the following formula:

$$\lambda_n = k_g\left(\frac{A}{V}\right)$$

where,

λ_n = removal rate constant due to surface deposition,

k_g = average mass transfer coefficient,

A = surface area for deposition, and

V = Volume of the contained gas.

The area used in the analysis was the wall surface area of the building, and the floor area for elevation 17500. Other surfaces, such as the bioshield wall for the drywell (above Elevation 17500), were conservatively neglected. The inside diameter of the drywell below elevation 17500 is 9292 mm:

$$A_{DW,<17500} = \pi DH = \pi(9.3m)(17.5m-[-10.0\ m]) = 803.5\ m^2$$

Only 50% of the area was credited to provide additional conservatism. The diameter of the drywell is 33.5 m, therefore,

$$A_{DW,17500} = 50\%*\pi r^2 = 0.5\pi(^{33.5m}/_2)^2 = 440.7\ m^2$$

$$A_{tot} = 803.5\ m^2 + 440.7\ m^2 = 1244.2\ m^2 = 13392.5\ ft^2$$

The removal rate constant assumed was 0.137 cm/sec (16.18 ft/hr) based on NUREG/CR-0009, Page 17; therefore,

$$\lambda_n = 16.18(^{ft}/_{hr})\left(\frac{1.34E4\,ft^2}{2.36E5\,ft^3}\right) = 0.92hr^{-1}.$$

No DCD changes are required as a result of this RAI.

DCD Impact:

DCD, Tier 2, Subsection 15.4.4.5.2.1, Revision 4 will include clarification on development of the of the iodine removal rate constant as noted on the attached markup.

NRC RAI 15.4-14:

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 LTR states that the amount of cable insulation material in the containment of the Advanced Boiler Water Reactor is applied to the ESBWR. Please include the amount of cable insulation material in the ESBWR containment in DCD Tier 1 as an ITAAC item.

GE Response:

DCD, Tier 1, Revision 3, Subsection 2.15.1 and Table 2.15.1-1 will be revised to include an ITAAC item for exposed cable mass as indicated on the attached markups.

DCD Impact:

DCD Tier 1, Subsection 2.15.1 and Table 2.15.1-1, Revision 4 will include an ITAAC item for exposed cable mass as indicated on the attached markup.

15B. LOCA INVENTORY

This appendix provides additional detail on the design basis core source term assumed in the Chapter 15 dose consequence analyses. The source term was calculated using the computer code ORIGEN2 (Reference 15B-1). The source term meets the requirements of Regulatory Guide 1.183, Section 3.1.

The design power level for the ESBWR is 4500 MWt for a core with 1132 shortened GE14 fuel bundles. Considering a licensing power 2% above the design level gives a total core power of 4590 MWt or a bundle average power level of 4.054 MWt/bundle. The core inventory for licensing basis evaluations is based on the GE14 bounding bundle inventory. This inventory is based on a bundle enrichment of 4.6% and a core average exposure of 35 GWD/MTU. Also, it assumes a power level of 5.75 MWt/bundle. A full length GE14 bundle was used with a uranium mass of 182 kg, rather than the shorter bundle for the ESBWR, hence the higher bundle power assumption. The GE14 full-length core inventory has been used for numerous power uprate licensing amendments. The linear heat generation rate is identical for both full length and ESBWR GE14 fuel. Also, when normalized to total length other parameters such as uranium mass are comparable. As such, use of a full-length bundle has a negligible impact on the overall source term, thus the results are appropriate for the ESBWR.

Table 15B-1 contains values applicable to the ESBWR for the 60 isotopes used by the NRC computer code RADTRAD (Reference 15B-2).

15B.1 References

15B-1 CCC-371, "RSICC Computer Code Collection – ORIGEN 2.1", Oak Ridge National Laboratory, May 1999.

15B-2 NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," USNRC, April 1998.

15.4.4.5.1.3 Reactor Power

The rated core thermal power of the ESBWR is 4500 MWth. Adding an additional 2% to account for instrument uncertainty (Regulatory Guide 1.49) yields a core thermal power for this analysis of 4590 MWth.

15.4.4.5.1.4 Iodine Chemical Distribution

RG 1.183, Appendix A, Section 2 states that "If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodine (CsI), 4.85% elemental iodine, and 0.15% organic iodine." Based on the application of the systems identified in Subsection 15.4.4.5.2.2 to control pH after 72 hrs, this chemical distribution for pH controlled pools is assumed in the analysis.

15.4.4.5.1.5 Radiation Decay and Daughter Products

The computer code RADTRAD allows users to track radiation decay for the duration of the event. It also has an option to account for the buildup of daughter products. Both options are used in this analysis.

15.4.4.5.2 Radionuclide Releases and Pathways

Two specific pathways are analyzed in releasing radionuclides to the environment: leakage from the primary containment building and leakage through the Main Steam Isolation Valves. Leakage through the MSIVs is not included in the containment leakage summation, as discussed in Section 6.2.6.3. The primary containment leakage pathway is assumed to be no greater than an equivalent release of 0.5% volume per day from the containment per plant Technical Specifications. The bulk of the primary containment leakage (98%, or 0.49% volume per day) is released into the Reactor Building. Reactor Building leaks to the environment at a rate specified in Table 15.4-5. The remaining 2% of primary containment leakage is assumed to leak though the PCCS into the airspace directly above the PCCS and IC pools. This leakage is quickly vented directly to the atmosphere. The final leakage pathway is MSIV leakage to the Turbine Building condenser. This pathway is discussed separately below.

15.4.4.5.2.1 Removal of Elemental Iodine from Containment

Natural deposition of elemental iodine is credited in the dose consequence analyses. The elemental iodine coefficient is based on guidance found in SRP 6.5.2. Specifically, the iodine removal rate constant for a particular compartment "n" is based on the following formula:

$$\lambda_n = k_g\left(\frac{A}{V}\right)$$

where,

λ_n = removal rate constant due to surface deposition (0.137 cm/sec based on NUREG/CR-0009 (Reference 15.4-11) page 17),

k_g = average mass transfer coefficient,

A = surface area for deposition, and

V = volume of the contained gas.

2.15.4 Passive Containment Cooling System

Design Description

The Passive Containment Cooling System (PCCS), in conjunction with the suppression pool, maintains the containment within its pressure limits for DBAs such as a LOCA, by condensing steam from the Drywell atmosphere and returning the condensed liquid to the Gravity Driven Cooling System (GDCS) pools. The system is entirely passive, with no moving parts. No action is required for the PCCS to begin operation.

The PCCS consists of six low pressure, independent trains, each containing a steam condenser (passive containment cooling condenser) that condenses steam on tube side and transfers heat to water in a large cooling pool (IC/PCC pool) located outside the primary containment, which is vented to atmosphere.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool. The IC/PCC pool subcompartments on each side of the reactor building communicate at their lower ends to enable full use of the collective water inventory, independent of the operational status of any given PCCS train.

Each train, which is open to the containment, contains a drain line to one of the three GDCS pool, and a vent discharge line the end of which is submerged in the pressure suppression pool.

The PCCS loops are driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA, and as such require no sensing, control, logic or power actuated devices for operation.

The PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.

Together with the suppression pool, the six PCC condensers limit containment pressure to less than its design pressure. The Dryer/Separator pool and Reactor Well shall have sufficient water volume to provide makeup water to the IC/PCC pools for the initial 72 hours of a LOCA.

The PCC condensers are closed-loop extensions of the containment pressure boundary. Therefore, there are no containment isolation valves and they are always in "ready standby."

The PCCS can be periodically pressure-tested as part of the overall containment pressure testing program. The PCC loops can be isolated for individual pressure testing during maintenance. The PCCS leakage limit is 0.02 L_a.

During refueling outages, in-service inspection (ISI) of PCC condensers can be performed, if necessary. Ultrasonic testing of tube-to-heater welds and eddy current testing of tubes can be done with PCCs in place.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.4-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Passive Containment Cooling System.

Table 2.15.4-1

ITAAC For The Passive Containment Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the PCCS is as shown in Figure 2.15.4-1. a. The PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.	1. Inspections of the as-built system will be conducted. a. Inspections of the as-built PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.	1. The as-built PCCS conforms to the basic configuration shown in Figure 2.15.4-1. a. The design reports of the as-built PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.
2. The ASME Code components of the PCCS retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those Code Components of the PCC System required to be hydrostatically tested.	2. The results of the hydrostatic test of the ASME Code Components of the PCC conform to the requirements in the ASME Code, Section III.
3. The PCCS heat exchangers have a leakage limit of 0.02 L_a.	3. A pneumatic test of the PCCS will be conducted as part of the pre-service containment integrated leak rate test.	3. Test report(s) document that leakage from the PCCS heat exchangers is $\leq$ 0.02 L_a.
4. The PCCS together with the pressure suppression containment system will limit containment pressure to less than its design pressure for 72 hours after a LOCA.	4. An analysis will be performed using similar or more conservative performance characteristics than those of a full-scale test unit of established performance capability.	4. Analyzed containment pressure at 72 hours after a LOCA is less than containment design pressure.

The surface area credited for deposition is wall surface area of the building and the floor area for elevation 17500, since that elevation represents the largest cross-section area. The resultant area (803.5 m^2) was then conservatively reduced by 50%. Other surfaces, such as the bioshield wall for the drywell are conservatively neglected. The calculated elemental iodine removal rate constant used is provided in Table 15.4-5.

15.4.4.5.2.2 Aerosol Removal from Containment

There are several natural processes which can remove airborne aerosols from the primary containment atmosphere following a LOCA. The PCCS is used to condense steam and control pressure in the event of a LOCA. The PCCS effectively scrubs the containment atmosphere by removing aerosols from the containment atmosphere. Aerosols are also removed via natural deposition onto containment internal structures. The removal mechanisms for the PCCS and natural deposition of airborne aerosols are similar, therefore one integral model is used. The removal coefficients are based on the results of the ESBWR MELCOR model as discussed in NEDE-33279P, *ESBWR Containment Fission Product Removal Evaluation Model* (Reference 15.4-13).

Aerosols released from the RPV will be airborne in the containment. One path for the aerosols to contribute to the off-site dose is via containment leakage from containment atmosphere to the Reactor Building and subsequently to the environment. Aerosols must be airborne in the containment atmosphere to leave via this pathway. Since aerosols are suspended in the containment atmosphere, they will circulate with the bulk gas movement, which is from the RPV to the PCCS. Figure 6.2-16 shows the PCCS heat exchanger and its associated piping. Steam, nitrogen and any airborne fission products will enter the PCCS heat exchanger inlet line from the drywell, which discharges to a header at the top of the PCCS tube bundles. Steam vapor will condense on the header and inside walls of the PCCS heat exchanger tubes, which are cooled on the outside by the water in the PCC/IC pool. The deposition processes of aerosol are gravity, Brownian diffusion, thermophoresis and diffusiopheresis. Aerosol and fission product vapors can deposit directly on surfaces such as heat structures and water pools. In addition, aerosol can agglomerate and settle. The aerosols deposited on the various surfaces can relocate. If a water film drains from a heat structure to the pool in the associated volumes, fission products deposited on that structure are transported with the water. This relocation is proportional to the fraction of the film that is drained. Aerosols and fission product vapors are transported between control volumes by bulk fluid and gas flows. Aerosols may also settle from a volume to a lower volume in the absence of bulk flow. Diffusiophoresis, the phenomenon of aerosol movement in condensing vapor, will drive the aerosol particles to the condensate film on the PCCS tube inner wall. In addition to condensation, some fraction of the airborne activity will also "plate out" in the PCCS. The PCCS effectiveness in removing aerosols has been demonstrated in third party tests. For example, in "Investigation on Aerosol Deposition in a Heat Exchanger Tube" (Reference 15.4-8), a short length of PCC tube was capable of removing a significant portion of in-flowing aerosols, and removing them with the condensate flow.

The condensate from the PCCS will drain into the GDCS pool, and then back into the reactor pressure vessel. The PCCS heat exchanger vents non-condensables, including noble gases to the suppression pool. Aerosols which do not deposit in the PCCS are transported by non-condensable gases via PCCS vent line into the wetwell. The vent mass flow rate is less than one-

Table 2.15.1-1

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The suppression pool water volume is equal to or greater than low water level volume used in the containment performance safety analysis.	7. Volumetric calculations will be performed using measured pool depth.	7. The calculated suppression pool water volume is equal to or greater than low water level volume used in the containment performance safety analysis.
8. The minimum suppression pool depth is 5.4 meters (17.7 ft).	8. Inspections of the containment suppression pool water level will be performed and measurements taken of the pool depth.	8. The minimum suppression pool depth is 5.4 meters (17.7 ft).
9. The vacuum breaker has proximity sensors to detect open/close position.	9. Inspections will be performed with the vacuum breakers in the open and closed positions, with indication of the open/close position.	9. Inspection report show that the vacuum breaker proximity sensors detect open/close position of vacuum breakers and provide indication.
10. Each vacuum breaker isolation valve automatically closes if the vacuum breaker does not fully close when required.	10. A test will be performed by simulating a not-fully closed vacuum breaker signal originating from the closed position proximity sensor.	10. Each vacuum breaker isolation valve automatically closes.
11. Control Room has indication of the open/close position for vacuum breakers.	11. Inspections will be performed on the Control Room for indicators of open/close position of vacuum breakers.	11.Indicators of open/close for vacuum breakers exist in the Control Room.
12. The amount of chlorine bearing cable insulation exposed to the containment atmosphere is limited.	12. Analyses and/or inspections will be used to confirm the final exposed chlorine bearing cable insulation mass.	12. The amount of chlorine bearing cable insulation exposed to the containment atmosphere (i.e., no enclosed cable tray, pipe conduit, etc.) is ≤ 3400 kg.

principal internal structure consists of the structural barrier separating the drywell from the wetwell. This barrier is comprised of the wetwell ceiling (diaphragm floor) and the inboard wall (vertical vent wall) separating the drywell from the wetwell. Both of these structural components are steel structures filled with concrete.

The containment structure and penetration isolation system with concurrent operation of other accident mitigation systems, are designed to limit fission product leakage during and following a postulated DBA to values well below leakage calculated for allowable off-site doses. The amount of chlorine bearing cable insulation exposed to the containment atmosphere (i.e., no enclosed cable tray, pipe conduit, etc.) that could produce chlorine following an accident is limited to ensure source term assumptions are maintained.

Vacuum relief between the drywell volumes and the wetwell gas space is provided by vacuum breakers. Each vacuum breaker is provided with an isolation valve in series, which automatically closes if the vacuum breaker fails to fully close when required. Each vacuum breaker has proximity sensors that provide position of open/close indication and an alarm in the main control room.

An all-steel reactor shield wall of appropriate thickness is provided, which surrounds the RPV to reduce gamma shine on drywell equipment during reactor operation and to protect personnel during shutdowns for maintenance and inservice inspections. The RPV insulation is supported from the internal surface of the reactor shield wall. The reactor shield wall is supported by the pedestal structure.

Table 2.15.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the Containment System.