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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 97 Related to ESBWR Design Certification Application – Safety Analyses – RAI Number 15.4-31**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated May 10, 2007. GEH response to RAI Number 15.4-31 is addressed in Enclosure 1.

If you have any questions or require additional information, please contact me.

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 07-292, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 97 Related To ESBWR Design Certification Application*, dated May 10, 2007.

Enclosure:

1. Response to Portion of NRC Request for Additional Information Letter No. 97 Related to ESBWR Design Certification Application – Safety Analyses – RAI Number 15.4-31

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRF	0000-0078-5272R1

Enclosure 1

MFN 08-023

Response to Portion of NRC Request for Additional Information Letter No. 97 Related to ESBWR Design Certification Application

Safety Analyses

RAI Number 15.4-31

NRC RAI 15.4-31:

Please state which set of control room X/Q values are used for the control room radiological consequences and why. In Revision 3 to the DCD, GE revised the control room X/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 X/Q values are provided for reactor building, passive containment cooling system/reactor building roof, and turbine building release pathways; one set for unfiltered in-leakage and the second set for the filtered air intake. Please state which set of control room X/Q values are used for the control room radiological consequences and why.

GEH Response:

For onsite radiological consequence evaluations using RADTRAD 3.03, X/Q values have been assumed for the Control Room unfiltered in-leakage locations and air intake locations (air intake is used for normal intake or emergency intake) and are based on the release location and nature of the release (either diffuse or point release). Since RADTRAD 3.03 allows only one set of X/Qs to be input for the Control Room, models with multiple release locations were analyzed separately using X/Q values for the release receptor pairs, and the results were then summed in order to determine the full impact. The full explanation of which set of CR X/Q values were used for each accident scenario evaluated in Section 15.4 of DCD Tier 2 Revision 4 is provided below.

It is important to note that for releases in the Turbine Building, X/Q values were calculated using ARCON96 based in part on conservative estimates of the distances between sources and receptors. In the case of the ARCON96 TB diffuse source model, the unfiltered in-leakage and normal air intake distances to the TB source were both conservatively reduced to 30 meters and both receptor locations are within the same wind window of 90 degrees. For that reason, the resulting X/Q values were identical.

1000 Fuel Rod Failure

For this scenario, it is postulated that fuel failure occurs in a number of fuel rods (1000 rods), releasing part of their fission product inventory. Two cases were considered for release paths to the environment. The first cases consists of the fission products traveling down the main steam lines, eventually reaching the offgas system, where they are held up in the charcoal delay beds and released to the environment through the main plant stack. For the second case, the offgas system is not available, and the fission products are transferred to the condenser, where they leak from the condenser into the Turbine Building without holdup and are subsequently released to the environment. The TB diffuse source release control room X/Q values were assumed to bound the offgas release from the main plant stack. Therefore, it was assumed that control room X/Q values used for the Turbine Building and offgas releases are the same, as follows:

Control Room X/Q Values (sec./m^3) for 1000 Fuel Rod Failure

Time	X/Q (sec/m^3)
0 – 2 hours	1.20E-03
2 – 8 hours	9.80E-04
8 – 24 hours	3.90E-04
1 – 4 days	3.80E-04
4 – 30 days	3.20E-04

It is important to note that the Control Room X/Q values for air intake for a release from the main plant stack were not included in DCD Revision 4, but will be included in Revision 5 of DCD Tier 1 Table 5.1-1 and Table 2.0-1 of the DCD Tier 2.

Liquid-Containing Tank Failure

The accident consists of the complete release of the radioactive inventory in all tanks containing radionuclides in the Liquid Waste Management System (LWMS). Due to mitigating capabilities of the Radwaste Building, it is assumed that there is no liquid release pathway to the environment. However, the airborne pathway is considered for this analysis. It is conservatively assumed that 100% of the iodine inventory of all the tanks in the LWMS is released through the airborne pathway to the environment.

The X/Q values used were those listed in DCD Table 2.0-1 for the Fuel Building Cask Door release to the control room air intake. Those values are assumed to bound any release from the Radwaste building due to their proximity to the Control Building.

Control Room X/Q Values (sec./m^3) for Liquid-Containing Tank Failure

Time Period	Radwaste Building
	CR Normal Air Intake
0 - 2 hrs	1.50E-03
2 - 8 hrs	1.30E-03
8 - 24 hrs	6.80E-04
1 - 4 days	5.60E-04
4 - 30 days	4.30E-04

It is important to note that the X/Q values for the Radwaste Building were not included in DCD Revision 4, but will be included in Revision 5 of DCD Tier 1 Table 5.1-1 and Table 2.0-1 of the DCD Tier 2.

FHA

The FHA discussed in DCD Tier 2 Revision 4, Section 15.4.1, is postulated to occur in Primary Containment or in the spent fuel pool in the Fuel Building. The three possible release locations identified for the FHA were a diffuse release from the Reactor Building, a point source release from the Fuel Building Cask Doors, and a diffuse release from the east side of the Fuel Building. The X/Q values used for the FHA evaluation were taken from DCD Tier 1 Table 5.1-1 and Tier 2 Table 2.0-1 for the Control Room Air Intakes and are presented in the table below. Because the Fuel Building Diffuse Source are bounding, these values were applied to the FHA analysis. Since normal ventilation is assumed, the dispersion factors were based on the Control Room Air Intake location.

Control Room X/Q Values (sec./m^3) for FHA

Time Period	Reactor Building – Diffuse Source	Fuel Building –Cask Doors	Fuel Building – Diffuse Source [2]
0 - 2 hrs	1.50E-03	1.50E-03	2.80E-03
2 - 8 hrs [1]	1.10E-03	1.30E-03	2.50E-03
8 - 24 hrs [1]	5.00E-04	6.80E-04	1.25E-03
1 - 4 days [1]	4.20E-04	5.60E-04	1.10E-03
4 - 30 days [1]	3.80E-04	4.30E-04	1.00E-03

Note [1] The release from the FHA is assumed to end after the first 2 hours, therefore, the X/Q values developed (given in DCD Table 2.0-1) for the time intervals after two hours were not used in the determination of dose consequences following an FHA.

Note [2] Since the Fuel Building – Diffuse Source values are bounding, these values are applied to the FHA analysis.

LOCA

For the LOCA analysis presented in DCD Tier 2 Revision 4, Section 15.4.4, three release locations were analyzed and the results summed in order to determine the radiological impact. The three release locations characterized were a diffuse release from the "east" side of the RB, point release from the PCCS, and a diffuse release from the TB via MSIV leakage there. Further, there were three accident scenarios investigated which accounted for simultaneous unfiltered in-leakage and air intake. As previously stated, RADTRAD allows only one set of X/Qs to be input for the Control Room. RADTRAD models for each release location were run and the results were summed. As such, the following X/Q values were all used in the LOCA dose evaluation in order to identify the bounding case.

Control Room X/Q Values (sec./m^3) for LOCA

Time	Reactor Building – Diffuse Source		PCCS/RB Roof (PCCS Leakage)		Turbine Building (MSIV Leakage)	
	Louvers	Emergency Intake	Louvers	Emergency Intake	Louvers	Emergency Intake
0 – 2 hrs	1.90E-03	1.50E-03	3.40E-03	3.00E-03	1.20E-03	1.20E-03
2 – 8 hrs	1.30E-03	1.10E-03	2.70E-03	2.50E-03	9.80E-04	9.80E-04
8 – 24 hrs	5.90E-04	5.00E-04	1.40E-03	1.20E-03	3.90E-04	3.90E-04
1 – 4 days	5.00E-04	4.20E-04	1.10E-03	9.00E-04	3.80E-04	3.80E-04
4 – 30 days	4.40E-04	3.80E-04	7.90E-04	7.00E-04	3.20E-04	3.20E-04

The Control Building louver X/Qs were used for modeling unfiltered in-leakage, and the emergency intake X/Qs were used for modeling the filtered intake. Due to geometric symmetry of the TB with respect to the Control Room intakes, the X/Q values used for MSIV leakage in the TB are identical for the Control Building louvers and emergency intakes.

MSLB

Two cases were considered for the postulated MSLB in DCD Tier 2 Revision 4, Section 15.4.5: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the activity was assumed to be released to the atmosphere as a diffuse release from the TB while the CR is operating in normal mode. The CR TB diffuse release X/Q values used are presented in the table below.

Control Room X/Q Values (sec./m^3) for MSLB

Time Period	X/Q (sec/m^3)
0 - 2 hrs	1.20E-03
2 - 8 hrs [1]	9.80E-04
8 - 24 hrs [1]	3.90E-04
1 - 4 days [1]	3.80E-04
4 - 30 days [1]	3.20E-04

Note [1] The release from the MSLB is assumed to end after the first 5.5 seconds; therefore, the X/Q values developed (given in DCD Table 2.0-1) for the time intervals after 2 hours were not used in the determination of dose consequences following an MSLB.

FWLB

As shown in DCD Tier 2 Revision 4, Subsection 15.4.7, the release due to a FWLB is assumed to be a diffuse release from turbine building exterior walls to the normal Control Room intake receptor location. The X/Q values used for the FWLB dose consequence analysis were the same as the X/Q values used for the MSLB (shown above). The FWLB dose assessment was modeled as an instantaneous release therefore; the X/Q values developed for the Turbine Building (given in DCD Table 2.0-1) for the time intervals beyond 2 hours were not used in the determination of dose consequences following the accident.

ILB

The small line break outside containment discussed in DCD Tier 2 Revision 4, Subsection 15.4.8 is postulated to be a circumferential rupture of an instrument line (ILB) that is connected to the primary coolant system, which occurs outside the drywell but inside the Reactor Building. Normal ventilation is assumed for the CR for the duration of the accident. The onsite X/Q values used for this analysis are based on a Reactor Building diffuse release for the Control Room air intakes (from DCD Tier 1 Table 5.1-1 and Tier 2 Table 2.0-1) and are presented in the table below.

Control Room X/Q Values (sec./m^3) for ILB

Time	X/Q Value (sec./m^3)
0 - 2 hrs	1.50E-03
2 - 8 hrs	1.10E-03
8 - 24 hrs	5.00E-04
1 - 4 days	4.20E-04
4 - 30 days	3.80E-04

RWCU

The postulated RWCU System line break in DCD Tier 2 Revision 4, Subsection 15.4.9 is postulated to occur in the Reactor Building and the release is assumed to occur as a diffuse source from the Reactor Building to the CR operating in normal mode. The X/Q values used in the radiological consequence evaluation of the RWCU are the same as those used for the ILB dose analysis (shown above). The RWCU System line break dose assessment was modeled as an instantaneous release therefore, the X/Q values developed for the Reactor Building – Diffuse Source (given in DCD Table 2.0-1) for the time intervals beyond 2 hours were not used in the determination of dose consequences following the accident.

Spent Fuel Cask Drop Accident

As stated in Subsection 15.4.10 of DCD Tier 2 Revision 4, "The fuel building design is such that a spent fuel cask drop height of 9.2 m, as specified in SRP 15.7.5, is not exceeded. This feature, along with administrative procedures limiting the travel range of the Fuel Building crane during cask handling activities, precludes damage of equipment or release of radioactivity due to dropping of a spent fuel shipping cask. Therefore, the radiological consequences of this accident are not evaluated."

Summary

The following Table summarizes the control room X/Q values used in the accident scenarios as well as the bases for these values:

DCD Section	Accident Scenario	Control Room X/Q Values Used from DCD Table 2.0-1	Bases for CR X/Q selection
15.4.1	Fuel Handling Accident	1. RB Diffuse Source for Air Intake 2. Fuel Building Diffuse Source [1]	The FHA is assumed to occur as a result of a failure of the fuel assembly lifting mechanism, resulting in dropping a raised fuel assembly onto the reactor core or into the spent fuel storage pool. Since the CR is assumed to operate in normal mode, the diffuse releases from the RB and FB were evaluated for the CR air intake.
15.4.4	Loss-of-Coolant Accident Inside Containment Radiological Analysis	1. RB Diffuse Source for Air Intake 2. RB Diffuse Source for Unfiltered In-leakage 3. PCCS Leakage for Air Intake 4. PCCS Leakage for Unfiltered In-leakage 5. TB Diffuse Source Air Intake and Unfiltered In-leakage [2]	For the LOCA analysis, five sets of X/Q values were examined in order to evaluate the three release locations considered (RB, PCCS, and TB via the MSIV) under the conditions of either unfiltered in-leakage or air intake (normal or emergency). Six possible release scenarios were identified and considered in order to determine the bounding dose consequence case.
15.4.5	Main Steamline Break Accident Outside of Containment	1. TB Diffuse Source Air Intake and Unfiltered In-leakage [2]	The X/Q values for the main control room are based on the turbine building release point. Due to symmetry, Turbine Building X/Q values are identical for both unfiltered in-leakage and air intakes.
15.4.6	Control Rod Drop Accident	N/A	As stated in DCD Tier 2 Revision 4, Section 15.4.6.3.1, "There is no technical basis for the control rod drop event to occur."
15.4.7	Feedwater Line Break Outside Containment	1. TB Diffuse Source Air Intake [2]	The release is assumed to be from the Turbine Building as a diffuse source through the Turbine Building walls. Since normal ventilation is assumed, the dispersion factors were based on the Control Room Air Intake location.

DCD Section	Accident Scenario	Control Room X/Q Values Used from DCD Table 2.0-1	Bases for CR X/Q selection
15.4.8	Small Line Break Outside Containment (Instrument Line Break)	1. RB Diffuse Source for Air Intake	The release is assumed to be from a Reactor Building as a diffuse source to the Control Room air intakes (normal ventilation is assumed for CR).
15.4.9	RWCU/SDC System Line Failure Outside Containment	1. RB Diffuse Source for Air Intake	RWCU System line break occurs in the Reactor Building and the release is assumed to be a diffuse release from the Reactor Building to the CR operating in normal mode.
15.4.10	Spent Fuel Cask Drop Accident	N/A	The fuel building design is such that a spent fuel cask drop height of 9.2 m, as specified in SRP 15.7.5, is not exceeded. This feature, along with administrative procedures limiting the travel range of the Fuel Building crane during cask handling activities, precludes damage of equipment or release of radioactivity due to dropping of a spent fuel shipping cask. Therefore, the radiological consequences of this accident are not evaluated.

[1]. The Control Room X/Q values for all releases from the FB have been determined for the CR air intakes

[2]. Due to symmetry, Turbine Building X/Q values are identical for unfiltered in-leakage and air intakes.

DCD Impact:

No DCD changes will be made in response to this RAI.