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Subject: **Response to Portion of NRC Request for Additional Information Letter Nos.181 Related to the ESBWR Design Certification – Safety Analyses – RAI Numbers 15.4-2S02, 15.4-3S02, 15.4-4S02, 15.4-6S01 through 15.4-8S01, 15.4-17S01, 15.4-22S01, 15.4-24S01 and 15.4-31S01.**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) responses to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated April 22, 2008 GEH responses to RAI Numbers 15.4-2S02, 15.4-3S02, 15.4-4S02, 15.4-6S01 through 15.4-8S01, 15.4-17S01, 15.4-22S01, 15.4-24S01 and 15.4-31S01 are addressed in Enclosure 1.

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

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

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NRO

Reference:

1. MFN 08-427, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 181 Related To ESBWR Design Certification Application*, dated April 22, 2008

Enclosure:

1. Response to Portion of NRC Request for Additional Information Letter No. 181 Related to ESBWR Design Certification Application – Safety Analyses – RAI Numbers 15.4-2S02, 15.4-3S02, 15.4-4S02, 15.4-6S01 through 15.4-8S01, 15.4-17S01, 15.4-22S01, 15.4-24S01 and 15.4-31S01

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRFs	0000-0085-6819 – RAIs 15.4-2S02, 15.4-3S02, 15.4-4S02 and 15.4-31S01
		0000-0086-1081 – Remainder of RAIs

Enclosure 1

MFN 08-467

Response to Portion of NRC Request for Additional Information Letter No. 181 Related to ESBWR Design Certification Application

Safety Analyses

RAI Numbers 15.4-2 S02, 15.4-3S02, 15.4-4S02, 15.4-6 S01, 15.4-7 S01, 15.4-8 S01, 15.4-17 S01, 15.4-22 S01, 15.4-24 S01, and 15.4-31 S01

NRC RAI 15.4-2 S02:

GEH should update reference to Main Steam Line diameter changes (from 24 inch to 30 inch outside diameter) in DCD.

GEH Response:

The Main Steam Line Break dose analysis has been revised as indicated in DCD Revision 5, including the design change of Main Steam Line to 30-inches. The previous analysis assumed mass releases of 43184 kg and 20369 kg for liquid and steam releases, respectively (reference GEH response to NRC RAI 15.4-2, Supplement 1 transmitted to the NRC via GEH letter MFN 07-100, Supplement 1 and 07-456 dated November 16, 2007). The revised analysis assumes mass releases of 45593 kg and 21084 kg for liquid and steam releases, respectively. These revised mass releases are based on a MSL diameter of 762 mm (30"), and DCD, Tier 2, Subsection 15.4.5, Revision 5 reflects the revised analysis based on the 762 mm Main Steam Line.

DCD Impact:

DCD Tier 2, Subsection 15.4.5, and Tables 15.4-11 through 15.4-13, were revised in DCD Revision 5. No additional changes are required.

NRC RAI 15.4-3S02:

a) Please add control room unfiltered air in-leakage rates to Tables 15.4-2 and 15.4-11 of DCD, Tier 2, Revision 4 and to Tables15.4-14 and 15.4-17 of DCD, Tier 2, Revision 5 markup copy.

b) Please recalculate control room doses using appropriate control room X/Q values provided in DCD Tier 2, Revision 4, Table 2.0-1, Envelope of ESBWR Standard Plant Design Parameters, for the normal air intake and unfiltered air intake for each respective design basis

c) For all design basis accident events where Control Room (CR) in-leakage value was assumed zero, GEH should increase the in-leakage to the value used in LOCA (12 cfm) and credit the Emergency Filtration Units (EFUs). This affects X/Q values for the unfiltered in-leakage path with the exception of the fuel handling accident (FHAs), which will have a higher X/Q value than LOCA, and will not credit the EFUs. GEH should re-calculate control room doses for all design basis accident events where Control Room (CR) in-leakage value was assumed to be zero.

d) For Feedwater Line Break (FWLB), GEH should add inleakage to CR.

e) For Instrument Line Break (ILB), GEH should revise Table 15.4-17 to include release point consistent with the format of all other accident results and input parameter tables.

GEH Response:

a). Control room (CR) unfiltered air in-leakage rates were added to Tables 15.4-2 (see '+' note), 15.4-11 (Item IIID), 15.4-14 (Item IIID) and 15.4-17 (Item IIID) in DCD, Tier 2, Revision 5.

b). The CR doses have been recalculated for the Section 15.4 events and can be found in DCD, Tier 2, Tables 15.4-2, 15.4-4, 15.4-5, 15.4-9, 15.4-11, 15.4-13, 15.4-14, 15.4-16, 15.4-17, 15.4-19, 15.4-21 and 15.4-23, Revision 5.

c). For the Fuel Handling Accident (FHA), the Control Room inleakage term was varied from 0 cfm to 10000 cfm, as indicated in DCD, Tier 2, Figure 15.4-4. The bounding values were 10000 cfm as indicated in DCD, Tier 2, Table 15.4-2. The Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Feedwater Line Break (FWLB), Small Line Carrying Primary Coolant Outside Containment, Reactor Water Cleanup/Shutdown Cooling Line Break CR in-leakage are all 12 cfm and are identified in Tables 15.4-5, 15.4-11, 15.4-14, 15.4-17 and 15.4-21 Item IIID, respectively. Revised doses associated with these changes for these events are provided in Tables 15.4-9, 15.4-13, 15.4-16, 15.4-19 and 15.4-23, respectively.

d). In-leakage for the FWLB is now identified in Item IIID of Table 15.4-14.

e). The release point for the instrument line break or Small Line Carrying Primary Coolant Outside Containment is now identified in as Item IE of Table 15.4-17.

DCD Impact:

DCD, Tier 2, Section 15.4, Revision 5 reflects the revised analyses discussed above. No additional changes are required.

NRC RAI 15.4-4S02:

a) Please add control room unfiltered air in-leakage rates to Tables 15.4-2 and 15.4-11 of DCD, Tier 2, Revision 4 and to Tables15.4-14 and 15.4-17 of DCD, Tier 2, Revision 5 markup copy.

b) Please recalculate control room doses using appropriate control room ?/Q values provided in DCD Tier 2, Revision 4, Table 2.0-1, Envelope of ESBWR Standard Plant Design Parameters, for the normal air intake and unfiltered air intake for each respective design basis accident.

c) For all design basis accident events where Control Room (CR) in-leakage value was assumed zero, GEH should increase the in-leakage to the value used in LOCA (12 cfm) and credit the Emergency Filtration Units (EFUs). This affects X/Q values for the unfiltered in-leakage path with the exception of the fuel handling accident (FHAs), which will have a higher X/Q value than LOCA, and will not credit the EFUs. GEH should re-calculate control room doses for all design basis accident events where Control Room (CR) in-leakage value was assumed to be zero.

d) For Feedwater Line Break (FWLB), GEH should add inleakage to CR.

e) For Instrument Line Break (ILB), GEH should revise Table 15.4-17 to include release point consistent with the format of all other accident results and input parameter tables.

GEH Response:

a). Control room (CR) unfiltered air in-leakage rates were added to Tables 15.4-2 (see '+' note), 15.4-11 (Item IIID), 15.4-14 (Item IIID) and 15.4-17 (Item IIID) in DCD, Tier 2, Revision 5.

b). The CR doses have been recalculated for the Section 15.4 events and can be found in DCD, Tier 2, Tables 15.4-2, 15.4-4, 15.4-5, 15.4-9, 15.4-11, 15.4-13, 15.4-14, 15.4-16, 15.4-17, 15.4-19, 15.4-21 and 15.4-23, Revision 5.

c). For the Fuel Handling Accident (FHA), the Control Room inleakage term was varied from 0 cfm to 10000 cfm, as indicated in DCD, Tier 2, Figure 15.4-4. The bounding values were 10000 cfm as indicated in DCD, Tier 2, Table 15.4-2. The Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Feedwater Line Break (FWLB), Small Line Carrying Primary Coolant Outside Containment, Reactor Water Cleanup/Shutdown Cooling Line Break CR in-leakage are all 12 cfm and are identified in Tables 15.4-5, 15.4-11, 15.4-14, 15.4-17 and 15.4-21 Item IIID, respectively. Revised doses associated with these changes for these events are provided in Tables 15.4-9, 15.4-13, 15.4-16, 15.4-19 and 15.4-23, respectively.

d). In-leakage for the FWLB is now identified in Item IIID of Table 15.4-14.

e). The release point for the instrument line break or Small Line Carrying Primary Coolant Outside Containment is now identified in as Item IE of Table 15.4-17.

DCD Impact:

DCD, Tier 2, Section 15.4, Revision 5 reflects the revised analyses discussed above. No additional changes are required.

NRC RAI 15.4-6 S01:

a) GEH has proposed to use AS1 as a bounding case rather than AS2 and should provide justification. GEH should reference NUREG-1465 and revise the AS-1 discussion and dose consequences in the DCD.

b) DCD Tier 2, Section 15.4.4.5.2.1 should refer to the LTR NEDE-33279P Revision 1, Section 4.3.1.

c) For DCD Tier 2, Section 15.4.4.5.4, verify that LTR NEDE-33279P is consistent with the DCD.

GEH Response:

(a) GEH has revised the particulate removal coefficients based on the revised MELCOR analyses. The "instantaneous" removal coefficients for each accident scenario were transmitted to the NRC via MFN 07-466 Supplement 2 dated April 15, 2008. GEH revised the LOCA dose calculation as documented in DCD, Tier 2, Subsection 15.4.4, Revision 5 using the revised removal coefficients. Previously, GEH has conservatively terminated the removal coefficients for removal of airborne aerosols at 12 hours. The revised analysis credits removal of airborne aerosols up to 24 hours based on the MELCOR results. Accident Scenario (AS) 1 is bounding with the new removal coefficients (based on 24 hours).

(b) DCD, Tier 2, Subsection 15.4.4.5.2.1, Revision 5 refers to NEDE-33279P, Section 4.3.1.

(c) NEDE-33279P is being revised to document the revised analysis. The revised Licensing Topical Report was transmitted via GEH letter MFN 06-205, Supplement 3 dated July 9, 2008. The revised LTR contains additional details of the dose analyses, including Accident Scenarios 2 and 3.

DCD Impact:

Revision 5 of DCD Tier 2, Subsection 15.4.4.5.2.1 incorporated this RAI. No additional changes are required.

Revision 2 of LTR NEDE-33279P was provided on July 9, 2008 via GEH letter MFN 06-205, Supplement 3.

NRC RAI 15.4-7 S01:

a) In DCD Tier 2, Table 15.4-5, GEH should add a footnote to explain the point after which no credit is taken for fission product removal (for AS-1, if limiting) and the basis for the cutoff.

b) In DCD Tier 2, Section 15.4.4.2.1, GEH should appropriately re-word this section based upon making AS-1 bounding.

NRC RAI 15.4-17 S01:

a) In DCD Tier 2, Section 15.4.4.2.1, GEH should move the first two sentences to Section 15.4.3. In Table 15.4-5, add a footnote (or other method) to explain the point after which no credit is taken for fission product removal (for AS-1 if limiting) and the basis for the cutoff.

b) In DCD Tier 2, Section 15.4.4.2.1 appropriately re-word this section based upon making AS-1 bounding.

GEH Response to RAIs 15.4-7S01 and 15.4-17S01:

(a) Footnotes were added to DCD, Tier 2, Table 15.4-5 in Revision 5 of the DCD to explain the timing and basis for terminating the removal coefficients (set to 0). Further, the first two sentences in Subsection 15.4.4.2.1 were relocated to Subsection 15.4.3 in DCD, Revision 5.

(b) DCD Tier 2, Subsection 15.4.4.2.1 has been revised documenting the assumptions used in the revised LOCA dose calculation (based on AS-1).

DCD Impact:

DCD Tier 2, Subsections 15.4.3, 15.4.4.2.1 and Table 15.4-5 (see * note) were revised in DCD, Revision 5 as a result of this RAI. No additional changes are required.

<u>NRC RAI 15.4-8 S01</u>:

a) GEH should emphasize in DCD Tier 2, Section 15.4.3, that it has been demonstrated in DCD Tier 2, Section 6.3 Emergency Core Cooling (ECCS) that no fuel melting occurs in a LOCA. The dose evaluation in Section 15.4.4 will need to be re-worded to invoke assumed fuel melting.

b) In DCD Tier 2, Section 15.4.4, GEH should include words from 10 CFR 52.47 evaluate accidents that could result in the release of significant quantities of radioactive fission products.

c) In DCD Tier 2, Section 15.4.4.2.1, GEH should delete both paragraphs in their entirety and include additional verbiage for the AS-1 LOCA scenario.

<u>GEH Response</u>:

(a) DCD Tier 2, Subsection 15.4.3 has been revised to document that no fuel melting occurs for the ECCS (Tier 2, Section 6.3) LOCA analyses. DCD Tier 2, Subsection 15.4.5 was revised (in DCD, Revision 5) to clearly document the fuel melt assumptions used in the LOCA dose calculations.

(b) DCD, Tier 2, Subsection 15.4.4, Revision 5 includes wording similar to 10 CFR 52.47:

> "In accordance with 10CFR 52.47 (a)(2), the evaluated event demonstrates that the ESBWR design reflects the extreme low probability for accidents that could result in the release of significant quantities of radioactive fission products. The fission product release assumed for this evaluation is based upon a hypothesized accident that is generally assumed to result in substantial meltdown of the core with subsequent release to containment of appreciable quantities of fission products.

(c) The DCD has been revised to clearly document the sequence of events for the bottom drain line break with automatic depressurization (AS-1). See DCD Tier 2 Subsection 15.4.4.2.1, Revision 5.

<u>DCD Impact</u>:

DCD Tier 2, Subsections 15.4.3, 15.4.4, and 15.4.4.2.1 were revised in DCD Revision 5 as a result of this RAI. No additional changes are required.

NRC RAI 15.4-22 S01:

a) In DCD Tier 2, Section 15.4.4.5.2.3, GEH should explain why the VTT report was not used in the analyses?

b) In DCD Tier 2, Section 15.4.4.5.1.1, reference 15.4-12, GEH should remove the reference to BWROG methodology, which is based on source term from TID-14844.

c) GEH should add a reference to fission product removal LTR NEDE-33279P for detailed analysis modeling.

d) In Table 4.3 of LTR NEDE-33279P, for the conversion of MSIV leakage to SCFM, what containment conditions (temperature) are used to convert? GEH should add a row to the table, such that both the RADTRAD volumetric flow rate, and flow rate at standard conditions (for TS) are provided in Chapter 15.

e) GEH should justify 20% air mixing in the condenser and an iodine removal rate of 99.5%. Include 20% air mixing in the condenser as an item to be verified in the ITAAC table.

GEH Response:

(a) Clarification was added to DCD Tier 2, Subsections 15.4.4.5.2.4 and 15.4.4.5.2.5, Revision 5 describing how the VTT report was applied with respect to MSIV leakage.

(b) The reference to the BWROG methodology was removed from Subsection 15.4.4.5.2.4 in Revision 5 to the DCD (no reference to the BWROG methodology was identified in 15.4.4.5.1.1).

(c) Reference to the fission product removal report Licensing Topical Report (LTR) NEDE-33279P was added to DCD Tier 2, Subsection 15.4.4.5.2.4, Revision 5 with respect to condenser modeling.

(d) DCD, Tier 2, Table 15.4-5, includes the MSIV leak rate in standard conditions (consistent with Tech. Spec. value), and also the converted values based on containment pressure and temperature.

(e) The removal rate in the main condenser of 99.3% assumed in the LOCA dose calculation is based on detailed modeling of the main steam lines, main steam line drain lines, and the main condenser. Details of the MELCOR model performed are contained in VTT-R-06771-07, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment – Part 3," Revision 2 which was transmitted to the NRC via GEH letter MFN 07-466, Supplement 1, dated March 31, 2008.

As discussed in the DCD and NEDE-33279P, leakage past the MSIVs is assumed to travel through the main steam lines and the main steam line drain lines into the main condenser. Only 20% of the main condenser is credited. The 20% mixing assumption in the main condenser is based on engineering judgment rather than a rigorous analysis, and is consistent with that approved for the GE ABWR design.

DCD Revision 5 contains a revised LOCA dose calculation. The MSIV contribution is shown in the markup to DCD, Tier 2, Table 15.4-9. The MSIV contribution is not significant (~2-10%). The primary reason for these relatively low dose contributions is the plateout in the main condenser, which was calculated using the computer code MELCOR as documented in the VTT "Part 3" report. The VTT report assumed a main condenser volume of 5929 m^3, a "horizontal plate" area of 418 m^2, and "horizontal cylinder" area of 2793 m^2. DCD, Tier 1, Subsection 2.11.7 and Table 2.11.7-1 were modified to include an ITAAC to verify the above parameters based on the final "as-built" condenser design.

DCD Impact:

DCD, Tier 1, Subsection 2.11.7 and Table 2.11.7-1 were revised in DCD, Revision 5 as a result of this RAI.

DCD Tier 2, Subsection 15.4.4.5.2.4 and Table 15.4-5 were revised in DCD, Revision 5 as a result of this RAI.

No additional changes to the DCD are required.

NRC RAI 15.4-24 S01:

a) GEH should replace or supplement DCD Tier 2, Tables 15.4-4 a-d (MELCOR Seq. of Events) with descriptive text.

b) In DCD Tier 2, Section 15.4.4.2.1, appropriately re-word this section based upon making AS-1 bounding.

GEH Response:

(a) Tables 15.4-4 a through 15.4-4 d, which were proposed to be added to DCD Tier 2 via RAI 15.4-17 (see MFN 07-523), were subsequently deleted from the DCD via this RAI. Additional information explaining the assumptions from the bounding accident scenario (AS-1: bottom drain line break with ADS operating) was added to DCD, Tier 2, Subsection 15.4.4.2.1, Revision 5.

(b) DCD, Tier 2, Subsection 15.4.4.2.1 was revised to reflect the AS-1 bounding accident.

DCD Impact:

DCD, Tier 2, Subsection 15.4.4.2.1 was updated in Revision 5. DCD, Tier 2, Tables 15.4-4a through 15.4-4d, which were added by GEH's response to RAI 15.4-17, were deleted as a result of this RAI. They are not included in DCD, Revision 5.

No additional changes to the DCD are required.

NRC RAI 15.4-31S01:

a) GEH should revise this response to clarify that the two dose calculations are provided for in Fuel Building and Reactor Building accidents.

b) GEH should explain why different atmospheric dispersion factors (X/Q) are used for Fuel Handling Accident (FHA).

GEH Response:

GEH has revised the prior response to RAI 15.4-31 as shown in the Enclosure. This revised response supersedes the prior response transmitted as MFN 08-023 (2/15/08). Items a) and b) of this supplemental RAI (supplement 1) are incorporated into this revised response.

For all ESBWR onsite radiological DBA consequence evaluations using RADTRAD 3.03, X/Q values have been assumed for the Control Room (CR) unfiltered in-leakage (assumed to enter the CR via the control building louvers) and air intake locations. 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 Sections 15.3 & 15.4 of DCD Tier 2 Revision 5 is provided below.

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 case 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 Turbine Building (TB) vent 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 TB without holdup and are subsequently released to the environment. The TB control room X/Q values were assumed to bound the offgas release from the Turbine Building vent stack. Therefore, the TB control room X/Q values for CR in-leakage and the CR emergency intakes were applied for the TB and offgas release scenarios.

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

Time	Turbine Building CR Inleakage	Turbine Building CR Intakes
0 – 2 hours	1.20E-03	1.20E-03
2 – 8 hours	9.80E-04	9.80E-04
8 – 24 hours	3.90E-04	3.90E-04
1 – 4 days	3.80E-04	3.80E-04
4 – 30 days	3.20E-04	3.20E-04

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) in the Radwaste Building. 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 PCCS vent release. Those values are assumed to bound any release from the Radwaste building due to their proximity to the Control Building, and have therefore been assigned to the Radwaste Building for this analysis. The Liquid-Containing Tank Failure dose assessment was modeled as an instantaneous release from the PCCS vents, therefore; the X/Q values developed for the PCCS (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.

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

Time Period	Radwaste Building CR Inleakage	Radwaste Building CR Intake
0 - 2 hrs	3.40E-03	3.00E-03
2 - 8 hrs [1]	2.70E-03	2.50E-03
8 - 24 hrs [1]	1.40E-03	1.20E-03
1 - 4 days [1]	1.10E-03	9.00E-04
4 - 30 days [1]	7.90E-04	7.00E-04

Note [1] The release from the LRWTA is modeled in RADTRAD as an instantaneous release, 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.

Fuel Handling Accident

The FHA discussed in DCD Tier 2 Revision 5, Subsection 15.4.1, is postulated to occur either in the Reactor Building or in the spent fuel pool in the Fuel Building. Since the refueling equipment in the Reactor Building is independent of the refueling equipment in the Fuel Building, two separate evaluations were performed. The RADTRAD results indicated the Fuel Building release scenario yielded the largest dose consequences. The X/Q values used for the FHA evaluations were taken from Revision 5 of the ESBWR DCD Tier 1 Table 5.1-1 and Tier 2 Table 2.0-1 for the Control Room Emergency Air Intake and Control Room Inleakage and are presented in the table below.

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

Time Period	Reactor Building CR Inleakage	Reactor Building CR Intakes	Fuel Building CR Inleakage	Fuel Building CR Intakes
0 - 2 hrs	1.90E-03	1.50E-03	2.80E-03	2.80E-03
2 - 8 hrs [1]	1.30E-03	1.10E-03	2.50E-03	2.50E-03
8 - 24 hrs [1]	5.90E-04	5.00E-04	1.25E-03	1.25E-03
1 - 4 days [1]	5.00E-04	4.20E-04	1.10E-03	1.10E-03
4 - 30 days [1]	4.40E-04	3.80E-04	1.00E-03	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.

Loss-of-Coolant Accident

For the LOCA analysis presented in DCD Tier 2 Revision 5, Subsection 15.4.4, three release pathways to the environment were analyzed and the results summed in order to determine the radiological impact.

- Reactor Building exfiltration
- PCCS Vent release
- Turbine Building release (MSIV and Feedwater isolation valve leakage)

The X/Q values used in the LOCA dose evaluation for the three pathways are presented in the table below.

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

Time	Reactor Building		PCCS/RB Roof		Turbine Building	
	CR Inleakage	CR Intakes	CR Inleakage	CR Intakes	CR Inleakage	CR Intakes
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

Main Steamline Break Accident

Two cases were considered for the postulated MSLB in DCD Tier 2 Revision 5, Subsection 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 from the TB while the CR is operating in emergency mode. The CR TB X/Q values used are presented in the table below.

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

Time Period	Turbine Building CR Inleakage	Turbine Building CR Intakes
0 - 2 hrs	1.20E-03	1.20E-03
2 - 8 hrs [1]	9.80E-04	9.80E-04
8 - 24 hrs [1]	3.90E-04	3.90E-04
1 - 4 days [1]	3.80E-04	3.80E-04
4 - 30 days [1]	3.20E-04	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.

Feedwater Line Break Accident

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 emergency 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 to the Turbine Building 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.

Failure of Small Line Carrying Primary Coolant Outside Containment Accident

The small line break outside containment discussed in DCD Tier 2 Revision 5, Subsection 15.4.8 is postulated to be a circumferential rupture of an instrument line that is connected to the primary coolant system, which occurs outside containment but inside the Reactor Building. Emergency 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 release for the Control Room emergency 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 Small Line Break Carrying Primary Coolant

Time	Reactor Building CR Inleakage	Reactor Building CR Intakes
0 - 2 hrs	1.90E-03	1.50E-03
2 - 8 hrs [1]	1.30E-03	1.10E-03
8 - 24 hrs [1]	5.90E-04	5.00E-04
1 - 4 days [1]	5.00E-04	4.20E-04
4 - 30 days [1]	4.40E-04	3.80E-04

Note [1] The release from the ILBA is modeled in RADTRAD as an instantaneous release, 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.

Reactor Water Cleanup (RWCU)/Shutdown Cooling (SDC) System Line Failure Outside Containment

The postulated RWCU/SDC System line break in DCD Tier 2 Revision 5, Subsection 15.4.9 is postulated to occur in the Reactor Building and the release is assumed to occur through the blowout panels on the Reactor Building roof to the CR operating in emergency mode. The X/Q values used in the radiological consequence evaluation were taken from DCD Tier 2 Revision 5, Table 2.0-1. The RWCU System line break dose assessment was modeled as an instantaneous release therefore, the X/Q values developed for the Reactor Building blowout panels (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.

Control Room X/Q Values (sec./m^3) for RWCU Line Break Outside Containment

Time	Blowout Panel CR Inleakage	Blowout Panel CR Intakes
0 - 2 hrs	7.00E-03	5.90E-03
2 - 8 hrs [1]	5.00E-03	4.70E-03
8 - 24 hrs [1]	2.10E-03	1.50E-03
1 - 4 days [1]	1.70E-03	1.10E-03
4 - 30 days [1]	1.50E-03	1.00E-03

Note [1] The release from the RWCU is modeled in RADTRAD as an instantaneous release, 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.

Summary

The following Table summarizes the control room X/Q values used in the accident scenarios as well as the reasoning.

Table 1 – Control Room X/Q Values Used from DCD Table 2.0-1 for ESBWR Radiological Consequence Analyses

DCD Subsection	Accident Scenario	Control Room X/Q Values Used from DCD Table 2.0-1	Reasoning for CR X/Q Selection
15.3.1	1000 Fuel Rod Failure	1. Turbine Building CR Inleakage X/Qs 2. Turbine Building CR Intakes X/Qs	Two possible releases were evaluated for the 1000 fuel Rod Failure. 1. Fission products traveling down the main steam lines to the offgas system, where they are subsequently released to the environment through the Turbine Building vent stack. 2. Fission products are released to the condenser, where they leak from the condenser into the TB and are subsequently released to the environment The TB control room X/Q values were assumed to bound the offgas release from the Turbine Building vent stack. Therefore, the TB control room X/Q values for CR inleakage and the CR emergency intakes were used for both release scenarios.
15.3.16	Liquid-Containing Tank Failure	1. PCCS/Reactor Building Roof CR Inleakage X/Qs. 2. PCCS/Reactor Building Roof CR Intakes X/Qs.	The X/Q values used were those listed in DCD Table 2.0-1 for the PCCS release. Those values are assumed to bound any release from the Radwaste building due to their proximity to the Control Building, and have therefore been assigned to the Radwaste Building for this analysis.
15.4.1	Fuel Handling Accident	1. Reactor Building CR Inleakage X/Qs 2. Reactor Building CR Intakes X/Qs 3. Fuel Building CR Inleakage X/Qs 4. Fuel Building CR Intakes X/Qs	The FHA discussed in DCD Tier 2 Revision 5, Section 15.4.1, is postulated to occur either in the Reactor Building or in the spent fuel pool in the Fuel Building.

DCD Subsection	Accident Scenario	Control Room X/Q Values Used from DCD Table 2.0-1	Reasoning for CR X/Q Selection
15.4.4	Loss-of-Coolant Accident Inside Containment Radiological Analysis	1. Reactor Building CR Inleakage X/Qs 2. Reactor Building CR Intakes X/Qs 3. Turbine Building CR Inleakage X/Qs 4. Turbine Building CR Intakes X/Qs 5. PCCS CR Inleakage X/Qs 6. PCCS CR Intakes X/Qs	For the LOCA analysis three of four release locations were analyzed using RADTRAD to determine the radiological impact. The three LOCA release locations evaluated with RADTRAD were releases from the RB, PCCS, and the TB via MSIV leakage there. Those results were then added to the dose consequences determined for the fourth pathway (leakage through the feedwater lines in the main steam tunnel) to determine the full impact of the LOCA.
15.4.5	Main Steamline Break Accident Outside of Containment	1. Turbine Building CR Inleakage X/Qs 2. Turbine Building CR Intakes X/Qs	For the MSLB the activity was assumed to be released to the atmosphere from the TB while the CR is operating in emergency mode.
15.4.7	Feedwater Line Break Outside Containment	1. Turbine Building CR Inleakage X/Qs 2. Turbine Building CR Intakes X/Qs	The FWLB dose assessment was modeled as an instantaneous release in the Turbine Building
15.4.8	Small Line Break Outside Containment (Instrument Line Break)	1. Reactor Building CR Inleakage X/Qs 2. Reactor Building CR Intakes X/Qs	The release is assumed to occur outside containment but inside the Reactor Building.
15.4.9	RWCU/SDC System Line Failure Outside Containment	1. Blowout Panel/RB Roof CR Inleakage X/Qs 2. Blowout Panel/RB Roof CR Intakes X/Qs	The release is assumed to occur outside containment but inside the Reactor Building and vent through the blowout panels on the RB roof.

DCD Impact

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