GE-Hitachi Nuclear Energy Americas LLC

James C. Kinsey
Project Manager, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

MFN 07-017, Supplement 1

Docket No. 52-010

August 22, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application – Safety Analyses – RAI Number 15.4-1S01**

Enclosure 1 contains GE-Hitachi Nuclear Energy Americas (GEH) response to the subject NRC RAIs transmitted via Reference 1. Enclosure 2 contains the DCD Markups associated with this response.

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
NRO

Reference:

1. MFN 06-381 – Letter from US Nuclear Regulatory Commission (NRC) to David H. Hinds, *Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application,* dated October 11, 2006

Enclosures:

1. Response to NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application – Safety Analyses, RAI Number 15.4-1S01
2. DCD Markups

cc: AE Cubbage	USNRC (with enclosures)
GB Stramback	GEH /San Jose (with enclosures)
RE Brown	GEH /Wilmington (with enclosures)
eDRF	0063-2924

Enclosure 1

MFN 07-017, Supplement 1

Response to Portion of NRC Request for Additional Information Letter No. 69

Related to ESBWR Design Certification Application

Safety Analyses

RAI Numbers 15.4-1 S01

NRC RAI 15.4-1 S01:

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 X/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 burnup?

(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 X/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 X/Q value of 1.0E-3 s/m3 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 X/Q values from 0 to 30 days.

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

GEH Response Item 1:

Radiological consequences are now provided for a fuel assembly drop in the Reactor Building as well as in the Fuel Building. DCD, Tier 2, Subsection 15.4.1.5 will be clarified in DCD Tier 2 Revision 5 to clarify that the Fuel Building is the bounding release point.

GEH Response Item 2:

No COL action is necessary inasmuch as neither the Reactor Building nor the Fuel Building is required to maintain building integrity during fuel handling operations. Release assumptions are consistent with RG 1.183, Appendix B, Section 4.2 that states:

> "The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period."

Footnote 2 of Appendix B to RG 1.183 states that "The term isolation is included here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods". Since the refueling floor is located in the Reactor Building, crediting containment integrity or closure is unnecessary. Building integrity is not assumed for the event. Therefore, Footnote 2 of Appendix B to RG 1.183 does not apply to the ESBWR design. No DCD changes are required.

GEH Response Item 3: DCD, Tier 2, Tables 2.0-1 and 15.4-2 were revised to document the X/Q values for all release points assumed in the Fuel Handling Accident (FHA). See attached DCD markups.

GEH Response Item 4: DCD, Tier 2, Revision 3, Subsection 15.4.1.2.1, "System Operation" currently states: "No credit is taken for the control room charcoal filter trains. Control room ventilation is assumed to operate in normal operation mode for the duration of the event."

GEH Response Item 5: The amount of iodine, noble gases and alkali metals has been added to DCD, Tier 2 as Table 15.4-3 (DCD Tier 2, Revision 3, Table 15.4-3 has been renumbered as 15.4-3a) as indicated on the attached markups. Table 15.4-3a does not include alkali metals due to the infinite decontamination factor. Footnotes have been added to Tables 6.3-1 and 6.3-11 to clarify that the requirements of Regulatory Guide 1.183, Footnote 11 apply to the ESBWR as indicated on the attached markups.

GEH Response Item 6: The radial peaking factor (RPF) value of 1.5 previously assumed for the FHA and the 1000-rod dose consequence analysis may not be conservative. Therefore, the FHA was revised assuming a higher RPF value of 1.7 as indicated in the attached DCD markups.

The RPF is the ratio of the bundle power to the core average bundle power. The 1000 rods assumed to fail to bound the various Infrequent Events may not result in the failure of the entire bundle; therefore, the peaking factor to be used in the 1000 rod dose consequence analysis will be based on the maximum allowable Linear Heat Generation Rate (LHGR). DCD, Tier 2, Table 6.3-1 provides the maximum permissible LHGR for the core at 13.4 kW/ft, and is based upon GE14E fuel. In consideration of potential new fuel designs, a new limit of 14.4 kW/ft will conservatively be used in the dose consequence analyses to bound future fuel designs. The active fuel length for the ESBWR is 10 feet or 3.05 meters. There are 1,132 bundles in the ESBWR core with 87.333 full-length rods in a GE14E fuel bundle. The peaking factor assumed in the 1000 rod dose consequence analysis is determined as follows:

$$N_{rods,core} = \left(1132\,{}^{bundles}\!/_{core}\right)\left(87.333\,{}^{rods}\!/_{bundle}\right) = 98861.3\,rods$$

$$LHGR_{coreavg} = \frac{4500MW}{(98861.3\,rods)\left(10\,{}^{ft}\!/_{rod}\right)} = 4.552\,{}^{kW}\!/_{ft}$$

$$PF_{13.4kW/ft} = \frac{13.4\,{}^{kW}\!/_{ft}}{4.552\,{}^{kW}\!/_{ft}} = 2.94$$

$$PF_{14.4kW/ft} = \frac{14.4\,{}^{kW}\!/_{ft}}{4.552\,{}^{kW}\!/_{ft}} = 3.16 \Rightarrow 3.2$$

Therefore, a "peaking factor" of 3.2 will be assumed in the 1000 rod dose consequence analysis. The revised analysis will be included in DCD, Tier 2, Section 15.3, Revision 5.

GEH Response Item 7: The table was revised to indicate curie units, correct typographical error and state total of fuel bundles used with DF of 200. It should be noted that the increased RPF of 1.7 is used in the development of the revised table (see discussion to Item 6 of this RAI Supplement).

RADTRAD Isotope Number	Isotope#	Core Conc. (Ci/MWt) [DCD, Tier 2, Table 15B-1]	Core Act. (Ci)	FHA Gap Activity @ 24hrs* (Ci)	FHA Gap Activity @ 24hrs* (MBq)	RADTRAD Rel to Env+ (Ci)	RADTRAD Rel to Env+ (MBq)
3	Kr-85	3.33E+02	1.53E+06	4.58E+02	1.70E+07	4.58E+02	1.69E+07
4	Kr-85m**	7.38E+03	3.39E+07	2.48E+02	9.19E+06	2.41E+02	8.93E+06
5	Kr-87	1.42E+04	6.54E+07	2.04E-02	7.57E+02	1.84E-02	6.81E+02
6	Kr-88	2.01E+04	9.20E+07	3.95E+01	1.46E+06	3.76E+01	1.39E+06
33	I-131	2.68E+04	1.23E+08	2.71E+04	1.00E+09	1.36E+02	5.02E+06
34	I-132	3.90E+04	1.79E+08	1.94E+01	7.18E+05	9.12E-02	3.37E+03
35	I-133	5.51E+04	2.53E+08	1.71E+04	6.32E+08	8.48E+01	3.14E+06
36	I-134	6.09E+04	2.79E+08	2.41E-04	8.91E+00	1.04E-06	3.84E-02
37	I-135	5.17E+04	2.37E+08	2.88E+03	1.07E+08	1.41E+01	5.21E+05
38	Xe-133	5.48E+04	2.51E+08	3.31E+04	1.22E+09	3.31E+04	1.22E+09
39	Xe-135	1.82E+04	8.34E+07	2.02E+03	7.46E+07	2.01E+03	7.42E+07

Note*: *The "plenum activity" values listed account for 2 bundles (of the 1132 bundles in the reactor core), gap fractions consistent with RG 1.183, Table 3, and a RPF=1.7. The values are not adjusted for any pool DF.*

Note **: *RG 1.183, Table 3 states that a gap fraction of 0.1 should be applied to Kr-85. The revised ESBWR FHA conservatively applied this value to Kr-85m as well.*

Note #: *Only the noble gas and iodine isotopes are included per NRC request.*

Note +: *The values listed account for a pool decontamination factor of 200 for iodines.*

<u>GEH Response Item 8</u>: DCD, Tier 2, Table 2.0-1 and Table 15.4-2 were revised to clearly indicate the control room X/Q values used in the analysis, including the Fuel Building Cask Door to Control Room Air Intake. See attached markup.

<u>GEH Response Item 9</u>: DCD, Tier 2, Subsection 15.4.1.5 was revised to clarify the bounding release pathway and why it is bounding. See attached DCD markups.

<u>GEH Response Item 10</u>:

a) DCD, Tier 2, Table 15.4-2 has been revised to clarify the Control Room X/Q values used in the FHA dose consequence analysis. The value of 1.0E-03 s/m^3 is not applicable to the revised analyses.

b) DCD, Tier 1, Subsection 2.16.2.2, "Control Building HVAC System," and Table 2.16.2-4, "ITAAC For The Control Building Habitability HVAC Subsystem," have been revised to include the control room air intake flow rate and the control room habitability area volume as indicated on the attached markup.

<u>GEH Response Item 11</u>: DCD, Tier 2, Table 15.4-4 was revised to delete the typographical error, recalculate the LPZ using the X/Q's for 0 to 30 days with an LPZ dose for 0 to 30 days. See attached DCD markups.

DCD Impact:

DCD, Tier 1, Section 2.16.2.2, "Control Building HVAC System," and Table 2.16.2-4, "ITAAC For The Control Building Habitability HVAC Subsystem," have been revised as noted on the attached markup and will be reflected in DCD, Tier 1, Revision 5.

DCD, Tier 2, Table 2.0-1, Table 6.3-1, Table 6.3-11, Subsection 15.4.1, Table 15.4-2, Table 15.4-3 (new table), Table 15.4-3a (previously Table 15.4-3), and Table 15.4-4 will be revised as noted on the attached markup and reflected in DCD Tier 2, Revision 5.

Enclosure 2

MFN 07-017, Supplement 1

DCD Markups

15.4 ANALYSIS OF ACCIDENTS

15.4.1 Fuel Handling Accident

15.4.1.1 Identification of Causes

The fuel-handling accident 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.

15.4.1.2 Sequence of Events and Systems Operation

Sequence of Events

The sequence of events is provided in Table 15.4-1.

Identification of Operator Actions

The following actions are carried out:

- Initiate the evacuation of the Reactor Building or Fuel Building fuel handling area and the locking of the fuel building doors;
- The fuel-handling foreman gives instructions to go immediately to the radiation protection decontamination area;
- The fuel-handling foreman makes the operations shift engineer aware of the accident;
- The shift engineer determines if the normal ventilation system has isolated;
- The shift engineer initiates action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the Reactor Building or Fuel Building;
- The duty shift engineer posts the appropriate radiological control signs at the entrance of the Reactor Building or Fuel Building; and
- Before entry to the fuel handling area is made, a careful study of conditions, radiation levels, etc., is performed.

15.4.1.2.1 System Operation

Normally operating plant instrumentation and controls are assumed to function. No credit is taken for the control room charcoal filter trains or the integrity of the Reactor Building or the Fuel Handling Building. Control room ventilation is assumed to operate in normal operation mode for the duration of the event. Operation of other plant reactor protection or engineered safety feature (ESF) systems is not expected.

15.4.1.3 Core and System Performance

15.4.1.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the radiological consequences of this accident are based on NUREG-1465 alternative source terms (AST) and the methodology in Regulatory Guide (RG) 1.183, to demonstrate compliance with the 10 CFR 50.34(a)(1), SRP 15.0.1 and RG 1.183 total effective ~~ed~~ dose equivalent (TEDE) acceptance criteria.

15.4.1.3.2 Input Parameters and Initial Conditions

Regulatory Guide (RG) 1.183 provides assumptions acceptable to the NRC that may be used in evaluating the radiological consequences of a postulated fuel-handling accident resulting in damage to the fuel cladding and subsequent release of radioactive materials.

15.4.1.3.3 Number of Failed Fuel Rods

The bounding event with respect to the number of fuel rods damaged occurs in the Reactor Building. Failure of the fuel rod is assumed at 1% circumferential strain. The associated axial strain is (.01)/ υ, where υ, Poisson's ratio, is 0.5 for plastic deformation, and thus the energy per rod failure is

$$Ef = \sigma y \times \varepsilon \times Vol$$

The kinetic energy of the dropped fuel bundle accounts for the effects of buoyancy and the resistance of water. Finite Element Analysis (FEA) simulations determined that when the drop distance of a fuel bundle is greater than 2.3 m (7.5 ft), the kinetic energy of the bundle is less than 50% in water than in air. When the bundle reaches a drop height of 10.36 m (34 ft), the energy is only ~22% ~~if~~ of that in air.

The fuel assembly wet weight is assumed to be 215 kg (474 lbs), and the mast wet weight is 195 kg (430 lbs). For conservatism in the analysis for an ESBWR (a drop height of 23.038 m [75.6 ft]), a factor of 2 reduction is applied to obtain the available energy in a fuel assembly drop through water. Therefore the kinetic energy as a result of the drop is

$$KE = (215kg + 195kg) \times (23.038m) \times 50\% = 4722.8kg - m$$

Half of the energy ~~ies~~ is assumed to be absorbed by the impacted assemblies. The ratio of the cladding to the non-fuel mass is 0.485. The calculated yield strength using the methodology described above is 35.515 kg-m/rod (256.88 ft-lb/rod). Therefore the number of failed rods from the initial drop is calculated as follows:

$$\frac{(50\%)(4722.8)(0.485)}{35.515\,{}^{kg-m}\!/_{rod}} = 32.25rods \Rightarrow 33rods$$

The fuel bundle is assumed to have a height of 3.6 m (141.7 in). One again accounting for a 50% reduction in water:

$$KE_2 = 50\% \times \left[h_{fuel} W_{mast} + \tfrac{1}{2} h_{fuel} W_{fuel} \right]$$

$$KE_2 = 0.5\left[(3.6m)(195) + \tfrac{1}{2}(3.6m)(215lb)\right] = 544.5kg - m$$

Once again 50% is assumed to be absorbed by the impacted assemblies, therefore the number of failed rods from the secondary impact is

$$\frac{(50\%)(544.5kg-m)(0.485)}{35.515^{kg-m}/_{rod}} = 3.7rods \Rightarrow 4rods$$

All of the 92 rods in the dropped assembly are assumed to fail, therefore the total number of rods (and bundles) failed are

$$92rods + 33rods + 4rods = 129rods$$

$$\left(\frac{129rods}{92^{rods}/_{bundle}}\right) = 1.4bundles \Rightarrow 2.0bundles$$

15.4.1.4 Radiological Consequences

Radiological analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR 50.34 and 10 CFR 50, Appendix A, General Design Criterion 19 guidelines.

The fission product inventory in the fuel rods that are assumed to be damaged is based on the days of continuous operation at full power. Due to plant cool down and disassembly operations, there is a time delay following initiation of reactor shutdown before fuel movement operations can be initiated. The analysis is based on Regulatory Guide 1.183. Specific values or parameters used in the evaluation are presented in Table 15.4-2.

15.4.1.4.1 Fission Product Transport to the Environment

~~The e~~Emergency procedures ~~protection guides~~ require that under FHA conditions the HVAC system be shut down and the fuel-handling area of the Reactor Building or Fuel Building isolated. Following isolation, the operator determines the extent of contamination and time for resuming operation of the HVAC. Gases are ~~assumed to be~~ released to the environment over a 2-hour period in accordance with Regulatory Guide 1.183 guidance. ~~This~~ The flow rate assumed ~~rate~~ in the dose consequence analysis exceeds the design flow rate for the Fuel Building ventilation system~~s~~ and the Reactor Building ventilation refueling floor subsystem (REPAVS).~~; however, isolation of the Reactor Building ventilation refueling floor subsystem (REPAVS) is required to ensure the 2-hour release assumption is conservative.~~ ~~In addition, t~~The Control Room ventilation is assumed to operate in normal mode. No credit is taken for Control Room emergency filter unit (EFU) mitigation ~~from the charcoal~~nor is the Reactor Building or Fuel Handling Building integrity assumed. The total activity released to the environment is presented in Table 15.4-3a.

15.4.1.4.2 Assumptions to be Confirmed by the COL Applicant

- All items required to be confirmed by the COL Applicant are discussed in Section 15.4.11.

15.4.1.5 Results

Calculations are performed for releases from both the Reactor Building and the Fuel Building. The results indicate that the Fuel Building release point is bounding due to the higher

atmospheric dispersion factor. The results of this analysis are presented in Tables 15.4-4 for both offsite and control room dose evaluations and are within 10 CFR 50.34(a)(1), 10 CFR50, Appendix A, GDC 19, and RG 1.183 regulatory guidelines.

15.4.2 Loss-of-Coolant Accident Containment Analysis

The containment performance analysis is provided within Section 6.2, and demonstrates that containment systems meet their design limits for all postulated design basis events.

15.4.3 Loss-of-Coolant Accident ECCS Performance Analysis

The emergency core cooling system (ECCS) performance analysis evaluates the full spectrum of pipe breaks, including the worst case of piping break inside containment. This analysis is provided within Section 6.3, and demonstrates compliance with the 10 CFR 50.46 ECCS acceptance criteria.

15.4.4 Loss-of-Coolant Accident Inside Containment Radiological Analysis

This event assumes a worst case of piping break inside containment. This event is in part based on the fact that the ECCS performance analysis demonstrates to what level that the 10 CFR 50.46 ECCS acceptance criteria are met, and that the containment analysis demonstrates that containment systems meet their design limits.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

The following analysis is based on NUREG-1465 alternative source terms (AST) and the methodology in Regulatory Guide (RG) 1.183, and demonstrates compliance with the 10 CFR 50.34(a)(1), SRP 15.0.1 and RG 1.183 total effective dose equivalent (TEDE) acceptance criteria.

15.4.4.1 Identification of Causes

There are no realistic, identifiable events that would result in a pipe break inside the containment of the magnitude required to cause a LOCA coincident with a Safe Shutdown Earthquake (SSE). The subject piping is of high quality, designed to nuclear construction industry codes and standards, and for seismic and environmental conditions. However, because such an accident provides an upper limit estimate for the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

The sequence of events associated with this accident is presented in Section 6.3 for ECCS performance and Section 6.2 for barrier (containment) performance.

Following the pipe break and scram, the MSIVs close on the reactor water low level trip signal (Level 2). Some moments later, the reactor low water (Level 1) signal initiates the ADS and GDCS. The core remains covered throughout the accident and there is no fuel damage.

Table 2.0-1

Envelope of ESBWR Standard Plant Site Design Parameters (continued)

Parameter	X/Q		
Meteorological Dispersion (X/Q): (11)	EAB X/Q:		
	0-2 hours:	2.00E-03 s/m^3	
	LPZ X/Q:		
	0-8 hours:	1.90E-04 s/m^3	
	8-24 hours:	1.40E-04 s/m^3	
	1-4 days:	7.50E-05 s/m^3	
	4-30 days:	3.00E-05 s/m^3	
	Control Room X/Q: *		
	Reactor Building – Diffuse Source		
	0-2 hours:	1.90E-03 s/m^3	1.50E-03 s/m^3
	2-8 hours:	1.30E-03 s/m^3	1.10E-03 s/m^3
	8-24 hours:	5.90E-04 s/m^3	5.00E-04 s/m^3
	1-4 days:	5.00E-04 s/m^3	4.20E-04 s/m^3
	4-30 days	4.40E-04 s/m^3	3.80E-04 s/m^3
	Passive Containment Cooling System / Reactor Building Roof		
	0-2 hours:	3.40E-03 s/m^3	3.00E-03 s/m^3
	2-8 hours:	2.70E-03 s/m^3	2.50E-03 s/m^3
	8-24 hours:	1.40E-03 s/m^3	1.20E-03 s/m^3
	1-4 days:	1.10E-03 s/m^3	9.00E-04 s/m^3
	4-30 days	7.90E-04 s/m^3	7.00E-04 s/m^3
	Turbine Building **		
	0-2 hours:	1.20E-03 s/m^3	1.20E-03 s/m^3
	2-8 hours:	9.80E-04 s/m^3	9.80E-04 s/m^3
	8-24 hours:	3.90E-04 s/m^3	3.90E-04 s/m^3
	1-4 days:	3.80E-04 s/m^3	3.80E-04 s/m^3
	4-30 days	3.20E-04 s/m^3	3.20E-04 s/m^3
	Fuel Building – Diffuse Source		
	0-2 hours:	NA	2.80E-03 s/m^3
	2-8 hours:	NA	2.50E-03 s/m^3
	8-24 hours:	NA	1.25E-03 s/m^3
	1-4 days:	NA	1.10E-03 s/m^3
	4-30 days:	NA	1.00E-03 s/m^3
	Fuel Building Cask Doors		
	0-2 hours:	NA	1.50E-03 s/m^3
	2-8 hours:	NA	1.30E-03 s/m^3
	8-24 hours:	NA	6.80E-04 s/m^3
	1-4 days:	NA	5.60E-04 s/m^3
	4-30 days:	NA	4.30E-04 s/m^3

—* First value is for unfiltered inleakage. Second value is for ~~filtered~~ air intakes (emergency and normal)

** Due to symmetry, Turbine Building X/Q values are identical for unfiltered inleakage and air intakes.

NA Values are not required for any dose analysis, therefore no values are available for the generic plant.

Table 2.0-1

Envelope of ESBWR Standard Plant Site Design Parameters (continued)

Meteorological Dispersion (X/Q): (11) **(continued)**	Radwaste Building		
	0-2 hours:	NA	1.50E-03 s/m^3
	2-8 hours:	NA	1.30E-03 s/m^3
	8-24 hours:	NA	6.80E-04 s/m^3
	1-4 days:	NA	5.60E-04 s/m^3
	4-30 days:	NA	4.30E-04 s/m^3
Long Term Dispersion Estimates: (12)	X/Q:	2.0E-06 s/m^3	
	D/Q:	4.0E-09 m^{-2}	

Table 2.16.2-4

ITAAC For The Control Building Habitability HVAC Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CRHAVS is as described in Subsection 2.16.2.2.	1. Inspections of the CRHAVS configuration will be conducted.	1. The as-built system conforms to the description in Subsection 2.16.2.2.
2. The CRHA isolation dampers automatically close upon receipt of a high radiation signal from PRMS, i.e. a. high radiation in the CRHAVS intake; b. high radiation downstream of an Emergency Filter Unit (EFU) during emergency operation, and c. low airflow through an EFU during emergency operation, or d. loss of AC power.	2. Using simulated high radiation isolation signals, tests will be performed on the (CRHA isolation dampers) isolation logic. A loss of AC power test will be performed.	2. Upon receipt of each simulated isolation signal; a. high radiation in the CRHAVS intake, b. high radiation downstream of the Emergency Filter Unit (EFU) during emergency operation, and c. low airflow through an EFU during emergency operation, or d. loss of AC power, the CRHA isolation dampers automatically close.
3. The safety-related components (EFUs, CRHA isolation dampers and associated components, instrumentation and controls) can withstand Seismic Category I loads without loss of safety-related function.	3. a. Type tests, analyses, or a combination of type tests and analyses of safety-related Seismic Category I equipment will be performed. b. Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the testing or analyzed conditions.	3. a. A report exists and concludes that the equipment can withstand seismic design basis without loss of safety-related function. b. A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The CRHAVS provides cooling to the CRHA.	4. Testing will be performed on the components using the controls in the MCR.	4. Controls in the MCR cause the components to perform the required function.
5. Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	5. a. Tests will be performed on CRHA isolation damper and EFU operation by providing a test signal in only one safety-related division at a time. b. Inspection of the as-installed safety-related divisions in the system will be performed.	5. a. The test signal exists only in the safety-related division under test in CRHA isolation damper and EFU control. b. Physical separation or electrical isolation exists between CRHA isolation dampers and EFU safety-related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.
6. Instrumentation showing the status of CRHA isolation damper and EFU operational status (Open/Closed) indication will be installed in the MCR.	6. a. Inspection will be performed to verify CRHA isolation damper and EFU operational status indication is installed in the MCR. b. Testing will be performed to show that the operational status indication in the MCR accurately depicts the operational status of the CRHA isolation dampers and EFUs.	6. a. The CRHA isolation damper and EFU operational status indication located in the MCR. b. A report exists and concludes that the operational status indication accurately depicts the operational status of the CRHA isolation dampers and EFUs.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Verify that the free air volume of the as-built control room envelope is greater than or equal to that assumed in safety analyses.	7. Analyses to be performed based on the final control room envelope design to determine free air volume (total volume minus equipment, walls, etc.) to be developed	7. Free air volume is greater than or equal to 2208 m^3 (78000 $ft.^3$).
8. Confirm normal operation intake flow rate.	8. Inspections will be performed to verify the actual flow rate.	8. Flow rate ≥ 200 l/s (424 cfm)

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

3 DPVs	sec	50
2 DPVs	sec	100
2 DPVs	sec	150
1 DPVs	sec	200
Total Number of Safety Relief Valves With ADS Function	—	10
Total Min. ADS Flow Capacity at Vessel Pressure	kg/hr MPa (gauge) [lbm/hr] [psig]	5.18×10^6 8.618 [11.4×10^6] [1250]
Total Number of Depressurization Valves	—	8
Total min. DPV flow capacity at vessel pressure	kg/hr MPa (gauge) [lbm/hr] [psig]	6.89×10^6 7.481 [15.2×10^6] [1085]
Total max. DPV flow capacity at vessel pressure	kg/hr MPa (gage) [lbm/hr] [psig]	8.47×10^6 7.481 [18.7×10^6] [1085]
C. Fuel Parameters *		
Variable	**Units**	**Value**
Fuel type	—	See Chapter 4
Peak Linear Heat Generation Rate~~*~~+ (Bounding)	kW/m [kW/ft]	44 [13.4]
Initial Minimum Critical Power Ratio (Bounding)	—	1.12

~~*~~+ LHGR limit for fuel burnups greater than 54 GWD/MT is 20.7 kW/m [6.3 kW/ft] peak rod average power.

Table 6.3-11

Plant Variables with Nominal and Bounding Calculation Values

Plant Variable	Nominal Value	Bounding Calculation Value*
1. Vessel Steam Dome Pressure	7.17 MPa (1040 psia)	7.274 MPa (1055 psia)
2. Decay Heat	1994 ANS (Figure 6.3-39)	+ 2σ
3. Core Power	Rated	+ 2%
4. PLHGR***	44.0 kW/m (13.4 kW/ft)	44.8 kW/m (13.7 kW/ft)
5. Initial MCPR	1.12	1.10
6. Initial Downcomer Level	NWL	NWL – 0.3m
7. Significant TRACG Modeling Parameters**	Nominal	Bounding

* Represents upper 95% or higher probability value.

** Reference 6.3-2, Table 2.5-2.

*** LHGR limit for fuel burnups greater than 54 GWD/MT is 20.7 kW/m [6.3 kW/ft] peak rod average power.

Table 15.4-2

FHA Parameters

I. Data and Assumptions Used to Estimate Source Terms		
A.	Power level, MWt	4590
B.	Core Source Term	Table 15B-1
~~B~~C.	Plenum Activity Radioactivity for I-131, % Radioactivity for Kr-85, % Radioactivity for other noble gases, % Radioactivity for other halogens, % Radioactivity for alkali metals, %	 8 10 5 5 12
~~C~~D.	Radial peaking factor for damaged rods	1.7~~5~~
~~D~~E.	Duration of accident, hr	2
F.	Total No. of Bundles in Core	1132
E.	No. bundles damaged	2
F.	Minimum time after shutdown to accident, hr	24
G.	Average fuel exposure, MWd/MT	35,000

II. Data and Assumptions Used to Estimate Activity Released	
A. Species fraction	
Released From Fuel	
Organic iodine, %	0.15
Elemental iodine, %	4.85
Particulate iodine, %	95
Noble gas, %	100
Reactor Building/Fuel Building Atmosphere	
Organic iodine, %	43
Elemental iodine, %	57
Particulate iodine, %	0
Noble gas, %	100
B. Pool Water Level, m (ft)	≥7.01 (23.0)
C. Pool Retention decontamination factor	

Iodine (effective)	200
Noble gas	1
Alkali metals/particulates	Infinite
C. Reactor Building release rate, %/hr	
0 – 1.95 hours	~~3~~500
1.95 – 2.0 hours	1.0E+08
III. Control Room Parameters	
A. Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E4)
B. Unfiltered intake, l/s (cfm)	200 (424)$^+$
C. Filtered intake, l/s (cfm)	0 (0)
D. Unfiltered inleakage, l/s (cfm)	0 (0)
E. Occupancy Factors	
0 – 1 day	1.0
1 – 4 days	0.6
4 – 30 days	0.4
III. Dispersion and Dose Data	
A. Atmospheric Dispersion Factors**	
~~Off-site~~Exclusion Area Boundary, sec./m^3	2.00E-03
Low Population Zone, sec./m^3	
0 – 8 hours	1.90E-04
8 hours – 30 days	N/A*
Control Room	
Reactor Building Release	~~Table 2.0-1~~
0 – 2 hours	1.50E-03
2 hours – 30 days	N/A*
Fuel Building Release	
0 – 2 hours	2.80E-03

	2 hours – 30 days	N/A*
B.	Dose conversion assumptions	RG 1.183
C.	Activity inventory/releases	Table 15.4-3
D.	Dose evaluations	Table 15.4-4

Note + - The design flow rate for the control room normal ventilation system is 200 l/s (424 cfm), however a value of 212.4 l/s (450 cfm) was conservatively used in the actual dose consequence analysis.

Note * - Since the release lasts only two hours, dispersion factors > 2 hours do not impact the calculated doses.

Note ** - See Table 2.0-1

Table 15.4-3
FHA Activity Released from Fuel

Isotope	(Ci)	(MBq)
I-131	2.71E+04	1.00E+09
I-132	1.94E+01	7.18E+05
I-133	1.71E+04	6.32E+08
I-134	2.41E-04	8.91E+00
I-135	2.88E+03	1.07E+08
Kr-85m	2.48E+02	9.19E+06
Kr-85	4.58E+02	1.70E+07
Kr-87	2.04E-02	7.57E+02
Kr-88	3.95E+01	1.46E+06
Xe-133	3.31E+04	1.22E+09
Xe-135	2.02E+03	7.46E+07
Cs-134	8.84E+03	3.27E+08
Cs-136	2.92E+03	1.08E+08
Cs-137	5.74E+03	2.12E+08
Rb-86	1.01E+02	3.74E+06

Table 15.4-3a

FHA Isotopic Release to Environment

Isotope	Activity (Ci)	Activity (MBq)
I-131	1.36E+02	5.02E+06
I-132	9.12E-02	3.37E+03
I-133	8.48E+01	3.14E+06
I-134	1.04E-06	3.84E-02
I-135	1.41E+01	5.21E+05
Kr-85m	2.41E+02	8.93E+06
Kr-85	4.58E+02	1.69E+07
Kr-87	1.84E-02	6.81E+02
Kr-88	3.76E+01	1.39E+06
Xe-133	3.31E+04	1.22E+09
Xe-135	2.01E+03	7.42E+07

Table 15.4-4

FHA Analysis Results

Accident Location, Exposure Location and Time Duration	Maximum Calculated TEDE (rem)	10 CFR 50.34(a)(1) Acceptance Criterion TEDE (rem)
~~Within Containment~~Reactor Building Release Results:		
[redacted] Exclusion Area Boundary (EAB) for ~~a~~ the worst 2 -hours ~~duration~~	4.13~~3.6~~	6.3
Outer boundary of Low Population Zone (LPZ) for ~~a 2-hour~~the duration of the accident (30 days)	0.39~~3.6~~	6.3
Control Room dose for the duration of the accident (30 days)	2.58~~2.3~~	5.0
Fuel Building Release Results:		
[redacted] Exclusion Area Boundary (EAB) for the worst 2 hours	4.13	6.3
Outer boundary of Low Population Zone (LPZ) for the duration of the accident (30 days)	0.39	6.3
Control Room dose for the duration of the accident (30 days)	4.82	5.0