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This letter forwards proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 1, the balance of this letter may be considered non-proprietary.

MFN 07-017

Docket No. 52-010

February 16, 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 – Safety Analysis – RAI Numbers 15.3-24, 15.3-25 and 15.4-1**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter. Enclosure 2 is a CD that contains RADTRAD input and output files requested by the NRC in RAI 15.3-25. Enclosure 2 contains GE proprietary information as defined by 10 CFR 2.390. GE customarily maintains this information in confidence and withholds it from public disclosure. Non-proprietary versions of the RADTRAD input and output files are not available and if prepared would effectively be empty files.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 2 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 2 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17. The Enclosure 2 computer files are entirely proprietary.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Project Manager, ESBWR Licensing

D068

References

1. MFN 06-381, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 69 Related to the ESBWR Design Certification Application*, October 11, 2006

Enclosures:

1. MFN 07-017– Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Numbers 15.3-24, 15.3-25 and 15.4-1
2. A CD titled MFN 07-017 Encl. 2, Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Number 15.3-25 - GE Proprietary Information
3. Affidavit – David Hinds – dated February 16, 2007

cc:	AE Cubbage	USNRC (with enclosures)
	David Hinds	GE/Wilmington (with enclosures)
	eDRF	0063-5563 for RAI 15.3-24
		0061-7874 for RAI 15.3-25
		0063-2924 for RAI 15.4-1

Enclosure 1

MFN 07-017

Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application

Safety Analysis

RAI Numbers 15.3-24, 15.3-25, 15.4-1

NRC RAI 15.3-24:

RAI Summary: Provide a more detailed discussion regarding the rad. assessments for IE.

Text: In DCD Tier 2, Rev. 1, Chapter 15, Table 15.3-1, the calculated delta CPR for the Infrequent Events varies from 0.0 to 0.15. Since the calculated delta CPR for all the events are small, according to the table, SLMCPR is not violated. However, radiological assessments are provided for several of them. Provide a more detailed discussion regarding the radiological assessment in spite of the low delta CPR. Also, specify the OLMCPR and the SLMCPR assumed in the analyses Subsection I.D given in the table is incorrect. Make the following editorial changes to the next DCD revision: 15.3.2.2 to 15.3.3 15.3.2.3 to 15.3.4 15.3.2.4 to 15.3.5 15.3.2.5 to 15.3.6 15.2.2.6 to 15.3.13 15.2.2.7 to 15.3.15

GE Response:

The bounding calculation to determine the off site dose for the infrequent events (IE) described in DCD Tier 2, Subsection 15.3.1.5 is based on a maximum number of 1000 rods in transition boiling which are assumed to fail. This off site dose bounding calculation resulted in off site doses lower than the acceptance criterion of 2.5 REM Total Effective Dose Equivalent (TEDE) and these results provide reasonable assurance that the dose evaluation will not be impacted by the future core designs (see DCD Tier 2, Table 15.3-16). However, due to uncertainties in the core design a note will be added to DCD Tier 2, Subsection 15.3.5.3.2 to check every reload core design for the number of rods in transition boiling such that the evaluation for the TEDE compliance remains acceptable.

The estimated number of rods in transition boiling in DCD Tier 2, Subsection 15.3.1 used in the radiological dose evaluation was determined to have a high degree of conservatism. The estimation of this number of failed rods was based on an assumed initial CPR of 1.28, (which is conservative relative to the OLMCPR, 1.3). This OLMCPR was determined such that 99.9% of the fuel rods avoid boiling transition during the transient of the limiting analyzed Anticipated Operational Occurrence. This is the ESBWR Safety Limit. There is no value for SLMCPR.

The determination of the rods in transition boiling was based on the maximum delta CPR found for the infrequent events shown in Table 15.3-1 (Load Rejection with total bypass failure, delta CPR equal to 0.16) and the initial CPR of 1.28 i.e. a chosen MCPR value of 1.12 (1.28 – 0.16).

GE response to RAI 15.3-9 provides a more detail discussion on the estimated bounding number of rods in transition boiling (failed) that is used in the IE radiological evaluation.

The subsection numbers identified in the "Subsection I.D." column of DCD Tier 2, Chapter 15, Table 15.3-1 will be revised to reflect the correct reference.

DCD Impact:

DCD Tier 2, Chapter 15 will be revised as noted below:

Subsection 15.0.2

FROM:

"The infrequent events considered in the ESBWR safety analyses are listed in Table 15.0-2, and are discussed in detail in Section 15.3. These include reactivity and power distribution anomalies such as the Control Rod Withdrawal Error events and the Loss of Feedwater Heating With Failure of Selected Control Rod Run-In event, which is re-evaluated for each fuel reload."

TO:

"The infrequent events considered in the ESBWR safety analyses are listed in Table 15.0-2, and are discussed in detail in Section 15.3. These consist of reactivity, power and pressure anomalies such as the Control Rod Withdrawal Error, the Loss of Feedwater Heating With Failure of Selected Control Rod Run-In and Generator Load rejection with Total Turbine Bypass Failure. The Loss of Feedwater Heating With Failure of Selected Control Rod Run-In and Generator Load rejection with Total Turbine Bypass will be re-evaluated for the specific initial core and reload core designs."

Subsection 15.3.5.3.2

FROM:

"This event will be analyzed for the plant-specific initial core configuration."

TO:

"This event is limiting with respect to the number of rods that enter transition boiling and will be analyzed for the plant-specific initial core configuration and subsequent reload core designs. The COL applicant will provide a reanalysis of this event for the specific initial core and (COL holder) reload core designs."

Table 15.3-1

The changes to Table 15.3-1 based on DCD Tier 2, Rev. 2 are shown below:

Third row	**From** 15.3.2 **to** 15.3.3
Sixth row	**From** 15.2.6 **to** 15.3.6
Seventh row	**From** 15.2.13 **to** 15.3.13
Eight row	**From** 15.2.15 **to** 15.3.15

NRC RAI 15.3-25:

Question Summary: Provide complete source term information for the radiological consequence analysis for certain infrequent events.

Full Text: A review of DCD Rev. 1, Tier 2, Section 15.3 has not provided complete source term information for the radiological consequence analysis for certain infrequent events. In Section 15.3, you listed 16 infrequent events. Out of these16 infrequent events, you have performed and provided the radiological consequence analyses for the following six infrequent events:Section 15.3.1Loss of Feedwater Heating with Failure of Selected Control Rod-In Section 15.3.4Pressure Regulator Failure - Closure of all Turbine Control and Bypass Valves Section 15.3.5Generator Load Rejection with Total Turbine Bypass Failure Section 15.3.6Turbine Trip with Total Turbine Bypass Failure Section 15.3.10 Fuel Assembly Loading Error - Misloaded Bundle Section 15.3.11 Fuel Assembly Loading Error - Misoriented Bundle Please provide the following additional source term information for the staff to perform an independent confirmatory dose calculation for the infrequent events listed above:

(A) Technical bases for assuming 1000 fuel rod failure with no fuel melt.

(B) Complete fission product inventory in reactor core at 4590 Mwt power level and state methodology used for developing the core inventory of fission products.

(C) Table 15.3-15 is titled as "1000 Fuel Rod Failure Core Fission Product Inventory." Are these fission product inventory in this table represent total fission product inventory in only 1000 failed fuel rods? Have you applied the radial peaking factor to these values?

(D) Condenser leak rate and duration of release from the condenser to the atmosphere.

(E) Amount of fission products released to the primary coolant from 1000 failed fuel rods.

(F) Amount of fission products reached the turbine and condenser.

(G) Amount of fission products released to the environment as function of time (for 0 to 2,2 to 8, and 8 to 24 hours)): (1) through condenser and (2) through off-gas system

(H) Technical bases for offgas dynamic adsorption coefficients and xenon holdup time in absorber beds used in dose calculation.

(I) Control room operator doses for these events

(J) Radiological consequence dose calculations performed for the Exclusion Area boundary (EAB), Low Population Zone (LPZ), and Control Room (CR). If an NRC computer code was used for the dose calculation (i.e., RADTRAD), please provide its input and output files.

GE Response:

Response to Item A

The value of 1000 rods is considered bounding for the most limiting transient event. A bounding analysis was performed for transient events, and it was determined that less than half of the assumed number of fuel rods would fail via cladding damage as a result of dryout. No fuel melt was postulated to occur during this limiting transient event. The value of 1000 rods was chosen to provide additional margin for the dose analysis that may be used to bound future fuel designs.

Response to Item B

The complete fission product inventory in the reactor core at 4590 MWth is provided in the table below:

	Isotope	Activity (MBq)		Isotope	Activity (MBq)		Isotope	Activity (MBq)
1	Co-58	2.34E+10	21	Ru-103	6.88E+12	41	Cs-136	3.16E+11
2	Co-60	2.26E+10	22	Ru-105	4.60E+12	42	Cs-137	5.89E+11
3	Kr-85	5.65E+10	23	Ru-106	2.39E+12	43	Ba-139	8.43E+12
4	Kr-85m	1.25E+12	24	Rh-105	4.18E+12	44	Ba-140	8.11E+12
5	Kr-87	2.42E+12	25	Sb-127	4.75E+11	45	La-140	8.35E+12
6	Kr-88	3.41E+12	26	Sb-129	1.45E+12	46	La-141	7.69E+12
7	Rb-86	1.08E+10	27	Te-127	4.82E+11	47	La-142	7.45E+12
8	Sr-89	4.56E+12	28	Te-127m	6.29E+10	48	Ce-141	7.70E+12
9	Sr-90	4.48E+11	29	Te-129	1.42E+12	49	Ce-143	7.18E+12
10	Sr-91	5.72E+12	30	Te-129m	2.11E+11	50	Ce-144	6.25E+12
11	Sr-92	6.15E+12	31	Te-131m	6.52E+11	51	Pr-143	7.02E+12
12	Y-90	4.76E+11	32	Te-132	6.48E+12	52	Nd-147	3.07E+12
13	Y-91	5.84E+12	33	I-131	4.55E+12	53	Np-239	8.87E+13
14	Y-92	6.18E+12	34	I-132	6.62E+12	54	Pu-238	1.54E+10
15	Y-93	7.09E+12	35	I-133	9.36E+12	55	Pu-239	1.84E+09
16	Zr-95	8.24E+12	36	I-134	1.03E+13	56	Pu-240	2.39E+09
17	Zr-97	8.48E+12	37	I-135	8.79E+12	57	Pu-241	6.95E+11
18	Nb-95	8.27E+12	38	Xe-133	9.30E+12	58	Am-241	7.82E+08
19	Mo-99	8.70E+12	39	Xe-135	3.09E+12	59	Cm-242	1.84E+11
20	Tc-99m	7.71E+12	40	Cs-134	9.08E+11	60	Cm-244	8.90E+09

The core inventory of fission products is based running ORIGEN2 for a fuel bundle with the following characteristics:

Fuel type:	GE14
Exposure:	35 GWd/MTU
Bundle enrichment:	4.6%
Bundle mass:	0.182 MTU
Bundle power:	5.75 MWth

Response to Item C

A radial peaking factor of 1.5 is used for this event. The 1000 rod fuel failure analysis has been revised. DCD Tier 2, Table 15.3-15 will be replaced with the table below (now called Table 15.3-14), which was generated in addressing item (E) of this RAI.

Table 15.3-14
1000 Fuel Rod Failure Fission Product Activity Released to Coolant

Isotope	Activity Released to Primary Coolant (MBq)
Kr-85	8.47E+07
Kr-85m	1.88E+09
Kr-87	3.63E+09
Kr-88	5.11E+09
Rb-86	1.94E+07
I-131	6.82E+09
I-132	9.92E+09
I-133	1.40E+10
I-134	1.55E+10
I-135	1.32E+10
Xe-133	1.40E+10
Xe-135	4.63E+09
Cs-134	1.63E+09
Cs-136	5.69E+08
Cs-137	1.06E+09

Response to Item D

The condenser leak rate is assumed to be 1% per day, in accordance with Regulatory Guide 1.183, Appendix C, Section 3.4. The duration of the release from the condenser to the atmosphere is 24 hours, in accordance with Regulatory Guide 1.183, Appendix C, Section 3.4.

Response to Item E

As discussed in Regulatory Guide 1.183, Appendix C, Section 1, 10% of the core inventory of the noble gases and iodine is in the fuel gap. Based on Table 3 of Regulatory Guide 1.183, 12% of the alkali metals (Cs, Rb) are in the fuel gap. Regulatory Guide 1.183, Appendix C, Section 3.1 states that the activity released from the fuel from the gap is instantaneously mixed in the reactor coolant within the pressure vessel. The revised 1000 rod fuel failure analysis calculates that 1000 rods is 1% of the number of total rods within the ESBWR core. As discussed in item (C) of this RAI response, a radial peaking factor of 1.5 is applied to the gap inventory. As a result, the amount of fission products released to the primary coolant from 1000 failed fuel rods is provided in the table below:

Isotope	Release to Primary Coolant (MBq)
Kr-85	8.47E+07
Kr-85m	1.88E+09
Kr-87	3.63E+09
Kr-88	5.11E+09
Rb-86	1.94E+07
I-131	6.82E+09
I-132	9.92E+09
I-133	1.40E+10
I-134	1.55E+10
I-135	1.32E+10
Xe-133	1.40E+10
Xe-135	4.63E+09
Cs-134	1.63E+09
Cs-136	5.69E+08
Cs-137	1.06E+09

Response to Item F

Per Regulatory Guide 1.183, Appendix C, Section 3.3, of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condenser. As a result, the amount of fission products released to the condenser from 1000 failed fuel rods is provided in the table below:

Isotope	Release to Condenser (MBq)
Kr-85	8.47E+07
Kr-85m	1.88E+09
Kr-87	3.63E+09
Kr-88	5.11E+09
Rb-86	1.94E+05
I-131	6.82E+08
I-132	9.92E+08
I-133	1.40E+09
I-134	1.55E+09
I-135	1.32E+09
Xe-133	1.40E+10
Xe-135	4.63E+09
Cs-134	1.63E+07
Cs-136	5.69E+06
Cs-137	1.06E+07

Response to Item G

Per Regulatory Guide 1.183, Appendix C, Section 3.4, of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. As a result, the amount of fission products from 1000 failed fuel rods available for release to the environment is provided in the table below:

Isotope	Available for Release from Condenser to Environment (MBq)
Kr-85	8.47E+07
Kr-85m	1.88E+09
Kr-87	3.63E+09
Kr-88	5.11E+09
Rb-86	1.94E+03
I-131	6.82E+07
I-132	9.92E+07
I-133	1.40E+08
I-134	1.55E+08
I-135	1.32E+08
Xe-133	1.40E+10
Xe-135	4.63E+09
Cs-134	1.63E+05
Cs-136	5.69E+04
Cs-137	1.06E+05

The amount (cumulative) of fission products released through the condenser to the environment as a function of time is provided in the table below, which will replace DCD Tier 2, Table 15.3-15:

Table 15.3-15
1000 Fuel Rod Failure Fission Product Activity
Cumulative Release to Environment

Isotope	Cumulative Release to Environment (MBq)		
	2 hours	8 hours	24 hours
Kr-85	7.07E+04	2.82E+05	8.44E+05
Kr-85m	1.34E+06	3.56E+06	4.77E+06
Kr-87	1.79E+06	2.65E+06	2.69E+06
Kr-88	3.33E+06	7.40E+06	8.49E+06
Rb-86	1.61E+00	6.42E+00	1.89E+01
I-131	5.67E+04	2.24E+05	6.49E+05
I-132	6.12E+04	1.23E+05	1.34E+05
I-133	1.13E+05	4.09E+05	9.42E+05
I-134	6.24E+04	7.84E+04	7.86E+04
I-135	9.85E+04	2.95E+05	4.63E+05
Xe-133	1.16E+07	4.54E+07	1.30E+08
Xe-135	3.57E+06	1.15E+07	2.05E+07
Cs-134	1.36E+02	5.44E+02	1.63E+03
Cs-136	4.73E+01	1.88E+02	5.50E+02
Cs-137	8.84E+01	3.53E+02	1.06E+03

In the case of the offgas system, it was assumed that all of the activity in the condenser is conservatively sent to the offgas system at once. The following offgas system parameters were used in the revised 1000 fuel rod failure analysis:

Xe holdup time:	60 days
k_{eff} (Kr):	0.0185 m^3/kg
k_{eff} (Xe):	0.3300 m^3/kg

The charcoal mass was calculated using the parameters above and the formula in Section 2.2.9.1 of NUREG-0016. This value was calculated to be 4.65E+05 lbm charcoal. The holdup time for the Kr isotopes was then calculated using this formula and the 4.65E+05 lbm charcoal mass. The Kr isotope holdup time was calculated to be 80.7 hours.

Using these holdup times and the noble gas activity available for release to the environment, it was calculated that the activity released from the offgas system is as follows:

Isotope	Available for Release from Offgas System to Environment (MBq)
Kr-85	8.56E+07
Kr-85m	7.16E+03
Kr-87	2.39E-10
Kr-88	1.08E+01
Xe-133	5.44E+06

The revised analysis conservatively assumed that all of the activity from the offgas system was instantaneously released to the environment.

Response to Item H

The dynamic adsorption coefficients for Kr (0.0185 m^3/kg) and Xe (0.3300 m^3/kg) are based on the current requirements of the offgas system design specification. These values are also consistent with the minimum values provided in Section 2.2.9.1 of NUREG-0016. The Xe holdup time is also based on the current requirements of the offgas system design specification.

Response to Item I

The control room operator doses are as follows:

Condenser release:	1.60E-01 rem TEDE
Offgas release:	8.56E-05 rem TEDE

Since the offgas release dose is significantly less than the condenser release dose (by three orders of magnitude), reference to the offgas release pathway will be eliminated from the ESBWR DCD. The only pathway that will be discussed in Revision 3 of DCD Tier 2 is the condenser leakage pathway.

Response to Item J

The following RADTRAD input and output files, as well as the nuclide inventory and release fraction (.nif and .rft, respectively) proprietary files are provided as an electronic attachment to this RAI response.

- ESBWR LOFWH_Condenser – No EBAS.psf

- ESBWR LOFWH_Condenser – No EBAS.o0
- ESBWR LOFWH_Condenser.nif
- ESBWR LOFWH_Condenser.rft

DCD Impact:

DCD Tier 2 Subsection 15.3.1.5, Tables 15.3-13, 15.3-14, 15.3-15, and 15.3-16 changes related to this accident are shown on the attached markup.

<u>NRC RAI 15.4-1</u>:

Question Summary: Provide source term assumptions for Fuel Handling Accident

Full Text: DCD Tier 2, Revision 1, Section 15.4.1, "Fuel Handling Accident," describes the postulated fuel handling accident and provides the resulting radiological consequence analysis. This section needs additional information regarding source term assumptions used by the applicant in order for the staff to perform an independent radiological consequence analysis.

(A) Please provide noble gases and iodine activity inventories in the fuel rod gaps (a) during normal operation at 4590 MWt with an average fuel burnup of 35 GWd/Mt, and (b) prior to fuel movement after 24 hour decay period that is available for release to the water surrounding the failed fuel assemblies. Also, please provide the amount of noble gases and iodine activities released to the environment following the postulated FHA.

(B) In DCD Tier 2, Revision 1, Table 15.4-2, "FHA Parameters," you provided specific values and parameters used in the postulated FHA analysis. Please provide technical bases for -21-assuming four failed fuel bundles due to the postulated FHA and state where this event is assumed to occur (i.e., inside containment, fuel handling area, auxiliary building, spent fuel pool).

(C) Please provide atmospheric dispersion factors (?/Q values) used for the EAB, LPZ,and CR.

(D) Please include in DCD Tier 2, Table 15.4-2 (1) the fraction of fission product in fuel gap used, (2) depth of water in the spent fuel pool that available for scrubbing fission product fission products before it is released, and (3) the release points (pathways) from the plant to the environment.

(E) If the FHA occurs in the containment, do you require closure of the containment purge lines? If you do require the closure, please state how you initiate the closure of the containment purge lines. If you rely on a radiation monitor to detect high airborne radioactivity, please state the sensitivity, range and setpoint of the radiation monitor. Is the ESBWR technical specifications require containment and/or Fuel Building closed during fuel movement, maintaining its integrity?

(F) In DCD Tier 2, Revision 1, Section 15.4.1.2 lists seven items of "Identification of Operator Actions" to be carried out by operators following postulated FHA. Please state any of these actions are subjected in the radiological consequence analysis, ESBWR technical specifications, and/or COL Action Items.

(G) The fuel and auxiliary pools cooling system for the spent fuel pool cooling and cleanup is a non-safety related system. Therefore, a loss of spent fuel pool (SFP) cooling capability should be analyzed coincident with the postulated FHA. The loss of SFP cooling could result in the pool reaching boiling, and a portion of the radioactive iodine in the SFP water could be released to the environment. Please provide the radiological consequence analysis for SFP boiling with a coincident loss of SFP cooling capability.

(H) Please state if 24 hour decay time, prior to movement of irradiated fuel, assumed in the -22-FHA analysis is specified in the ESBWR Technical Specification as a limiting condition for operation (LCO).

(I)Please provide complete FHA radiological consequence dose calculations performed for the EAB, LPZ, and CR. If an NRC computer code was used for the dose calculation (i.e.,RADTRAD, HABIT), please provide its input and output files.

(J) In DCD Tier 2, Revision 1, Section 15.4.1.4.2 lists "Assumptions to be Confirmed by the COL Applicant." Please state if these items will be specified as COL Action Items and/or Inspection, Test, Analysis and Acceptance Criteria (ITAAC) items.

(K) Please provide basis for reactor building release rate assumed as 350 percent per day.

(L) Please confirm the FHA isotopic release values to the environment provided in DCD Tier 2, Revision 1, Table 15.4-3, "FHA Isotopic Release to Environment," are correct.

GE Response:

General Information Concerning the FHA: The FHA dose consequence analysis documented in DCD Section 15.4 was revised by GE to address several issues:

- **Number of Failed Rods/Bundles:** The analysis prepared in support of DCD, Tier 2, Revision 0 (which is consistent with the analysis documented in Revision 2) conservatively assumed 4.0 bundles were damaged as a result of the drop. That value was based on a conservative assumption based on previous analyses prepared for the ABWR. The revised FHA dose consequence analyses utilized a drop with ESBWR specific parameters. The results indicated that the number of bundles damaged is 2.0.
- **Control Room Parameters:** The DCD, Tier 2, Revision 0 analysis assumed operation of the non-safety related charcoal filter train. Potential customers have requested that GE perform the analysis without crediting either the safety-related Emergency Breathing Air System (EBAS) or the non-safety related charcoal filter trains.
- **Atmospheric Dispersion Factors:** GE has performed additional research to determine atmospheric dispersion factors (X/Q values) which bound the near term ESBWR sites.
- **FHA Drop Locations:** The previous analysis only assumed a drop in the Reactor Building. The revised analysis reviews potential drops in both the Reactor Building (i.e., refueling pool) and the Fuel Building (i.e., the spent fuel pool).

The responses to RAI 15.4-1 documented below include discussions for the DCD, Tier 2, Revision 0 FHA dose consequence calculation, and the revised calculation as well. Unless otherwise noted all quantitative responses are based on the revised dose calculation. DCD markups for the FHA are included in this response.

Response to Item (A)

The gap fractions assumed in the ESBWR FHA dose consequence analysis are consistent with Regulatory Guide 1.183, Table 3. The requested data is presented in the Table that follows.

RADTRAD Isotope #	Isotope***	Core Concentration [t = 0 hrs] (Ci/MWt)	FHA Gap Activity* [t = 0 hrs] (Ci)	FHA Gap Activity* [t = 24 hrs] (Ci)	FHA Release to Environment [RADTRAD Results] (Ci)
3	Kr-85	3.33E+02	4.04E+02	4.04E+02	4.04E+02
4	Kr-85m**	7.38E+03	8.98E+03	2.19E+02	2.10E+02
5	Kr-87	1.42E+04	8.66E+03	1.80E-02	1.56E-02
6	Kr-88	2.01E+04	1.22E+04	3.49E+01	3.26E+01
33	I-131	2.68E+04	2.60E+04	2.39E+04	1.19E+02
34	I-132	3.90E+04	2.37E+04	1.71E+01	7.84E-02
35	I-133	5.51E+04	3.35E+04	1.51E+04	7.48E+01
36	I-134	6.09E+04	3.70E+04	2.12E-04	8.63E-07
37	I-135	5.17E+04	3.15E+04	2.54E+03	1.23E+01
38	Xe-133	5.48E+04	3.33E+04	2.92E+04	2.91E+04
39	Xe-135	1.82E+04	1.11E+04	1.78E+03	1.77E+03

Note*: *The "plenum activity" values listed account for 2 bundles, gap fractions consistent with RG 1.183, Table 3, and a RFP=1.5. 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 that value to Kr-85m as well.*

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

Response to Item B

The FHA analysis documented in DCD, Revision 1 assumed that 4 bundles were damaged as a result of the drop. This value was a conservative estimate based on other plant designs (specifically the Lungmen [ABWR] FHA analysis). GE has since performed a more rigorous analysis to calculate the number of fuel rods damaged with ESBWR design information. This more rigorous analysis is discussed below.

The bounding event with respect to the number of fuel rods damaged occurs in the Reactor Building. Previous analyses assumed fuel failure occurred at 1% axial compression strain. However, failure by 1% compressive strain is not a realistic failure mode. The associated tensile circumferential strain is a more realistic failure mode. Assuming failure 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

$$E_f = \sigma_y \times \varepsilon \times Vol = (86{,}000\ psi) \times (.01)/(0.5) \times \pi/4(OD^2 - (ID+2\times liner\ thickness)^2)(Tube\ length)/2$$

Note in this equation that the liner thickness is excluded, as the yield strength of the liner is less than that of the Zicaloy-2 portion of the clad.

Secondly, previous methodology did not take credit for the friction between the water and the fuel assembly in calculating the total energy available to fail fuel rods. New developments in simulations of fluid-structure interference interface using computational fluid dynamics (CFD) and finite element analyses (FEA) make it possible to calculate the velocity and energy of a fuel bundle drop in water. Simulations determined that when the drop distance of a fuel bundle is greater than 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 34 ft, the energy is only ~22% if that in air.

The fuel assembly wet weight is assumed to be 474 lbs, and the mast wet weight is 430 lbs. For conservatism in the analysis for an ESBWR (a drop height of 23.038 m [75.6 ft]), a 50% 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

$$KE = (474lb + 430lb) \times (75.58ft) \times 50\% \approx 34164 ft - lbs.$$

Half of the energy 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 256.88 ft-lb/rod. Therefore the number of failed rods from the initial drop is calculated as follows:

$$\frac{(50\%)(34164ft-lb)(0.485)}{256.88 \,{}^{ft\text{-}lb}/_{rod}} = 32.25rods \Rightarrow 33rods$$

The fuel bundle is assumed to have a height of 141.7 in. Once again accounting for a 50% reduction in water:

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

$$KE_2 = 0.5\left[(141.7in)(430lb) + \tfrac{1}{2}(141.7in)(474lb)\right]\left({}^{1ft}/_{12in}\right) = 3938ft-lb$$

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

$$\frac{(50\%)(3938ft-lb)(0.485)}{256.88 \,{}^{ft\text{-}lb}/_{rod}} = 3.7rods \Rightarrow 4rods$$

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

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

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

As a result, 2 bundles are assumed to fail in the revised FHA analysis for DCD, Revision 3.

<u>Response to Item C</u>

The X/Q value assumed for the EAB is 2.00E-03 sec./m^3, consistent with the revised LOCA dose calculation documented in NEDE-33279P. This value is assumed to be applicable to any release location for the ESBWR. COL applicants must verify their site is bounded by this value. Note that the "maximum" X/Q value for the FHA was also calculated for the EAB. This maximum value represents the largest value that would still result in dose consequences less than the Regulatory Limit of 6.3 REM TEDE (25% of 10 CFR 50.34 (a) 1). The "maximum X/Q" value was determined to be 3.47E-03 sec./m^3.

Since the release duration is short (2 hrs), a detailed calculation was not performed for the LPZ since it is bounded by the EAB results. As such, the LPZ X/Q is inherently assumed to be ≤2.00E-03 sec./m^3.

Two release locations were considered for a FHA: the reactor building (refueling pool) and the fuel building (spent fuel pool). Dispersion factors are available for the Reactor Building and the Cask Doors on the Fuel Building. Details for the calculations were provided previously to the NRC via GE's response to RAI 2.3-9. As discussed previously, the revised FHA dose consequence analysis assumes the Control Room remains in normal operation mode for the duration of the event. The bounding release location was determined to be the Fuel Building Cask Doors.

Time Period	Reactor Building to Control Room Air Intake	Fuel Building Cask Doors to Control Room Air Intake
0 - 2 hrs	1.50E-03 sec./m^3	1.50E-03 sec./m^3
2 - 8 hrs	1.10E-03 sec./m^3	1.30E-03 sec./m^3
8 - 24 hrs	5.00E-04 sec./m^3	6.80E-04 sec./m^3
1 - 4 days	4.20E-04 sec./m^3	5.60E-04 sec./m^3
4 - 30 days	3.80E-04 sec./m^3	4.30E-04 sec./m^3

Response to Item D

The revision to Table 15.4-2 includes the requested changes.

Response to Item E

The revised FHA dose consequence analysis reviews releases to both the Reactor Building and the Fuel Building. Any fission products released as a result of a drop in the refueling pool would be released into the Reactor Building atmosphere, specifically the refueling floor elevation (El. 34000 mm). Fission products released as a result of a drop in the Fuel Building are assumed to be released directly to the environment via the cask doors on the west side of the building. The release is assumed to occur over a 2 hour period consistent with Regulatory Guide 1.183, therefore there are no explicit assumptions with respect to building integrity. The analysis is not dependent on closure of containment purge lines or detection of high airborne radiation. The ESBWR does not have any Technical Specifications related to requiring building integrity during fuel movement.

Response to Item F

None of the operator actions listed in DCD Tier 2, Revision 1, Subsection 15.4.1.2 are credited in the FHA dose consequence analyses, therefore there are no Technical Specifications or COL Action Items tied to these actions.

Response to Item G

The FHA dose consequence analysis was performed in accordance with the requirements of Regulatory Guide 1.183, Appendix B. Regulatory Guide 1.183 does not require an assumed failure of spent fuel pool cooling coincident with a FHA.

NUREG-0800, U.S. NRC Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," was also consulted to evaluate the dose consequences from a FHA (both Revision 1 and DRAFT Revision 2). The SRP discusses requirements with respect to ESF filter trains, however no requirement with respect to spent fuel pool cooling was identified.

Although no licensing requirement was identified to evaluate the impact of pool boiling during a FHA, GE reviewed the pool heat loads during refueling operations. Design basis heat loads for the spent fuel pool (SFP) are based on the assumption that it takes 10 days to off-load the core to the SFP. The resultant heat load in the SFP is significantly lower than that of the core 24 hours after shutdown (when the FHA is assumed to occur), therefore the bounding event (with respect to potential pool boiling) is in the Reactor Building.

The heat load of the core at 24 hours is 25.0 MW. A simple calculation determined that it would take ~6 hours for the bulk temperature of the refueling pool and RPV to reach boiling. This evaluation conservatively neglected the water volumes in the dryer/separator storage pool, the spent fuel storage area, and the upper IFTS pool. It also excluded any heat transfer to internal surfaces.

There are numerous conservatisms in the FHA dose consequence analysis, several of which are discussed below:

- <u>Number of Bundles:</u> The number of bundles assumed to be damaged was 2.0. Since there are 87.33 equivalent full length rods per bundle, this equates to 174.66 equivalent full length rods. Item (B) discussed previously demonstrated that the calculated number of rods damaged was 129 EFLR. As such, there is ~35% in the source term assumed.
- <u>Number of Damaged Rods:</u> The calculated number of rods assumed a 50% reduction in kinetic energy as result of the drop in water. Detailed CFD calculations determined that the KE would be ~22% of the energy in air for a drop of 34 ft.
- <u>Decay Time:</u> A decay time of 24 hours is assumed consistent with the ESBWR Technical Specifications. In reality, it would be extremely difficult for fuel movement to commence as early as 24 hours. Numerous activities are required to occur prior to moving fuel. The RPV must be depressurized, RVP head bolts detensioned, head removed, dryer removed, moisture separator removed, main steam line plugs installed, and RPV vessel and refueling pools reflooded. These actions will likely take significantly longer than 24 hours to complete. Therefore, the decay time assumed in the dose analysis is extremely conservative. Note also that the decay heat assumed in the simple calculation was based on 24 hours. There is a ~20% reduction in decay heat from 24 to 48 hours, which would also increase the "time to boil" of the pool.

No licensing requirement was identified which requires a loss of FAPCS coincident with a FHA. Additional review confirms that there is considerable margin in the revised FHA dose consequence analysis. Also, using conservative assumptions it was determined that it would take ~6 hours for the RPV/refueling pool to reach bulk boiling, and due to the significant water mass involved the boil-off rate would not be significant. As such plant operators would have a significant amount of time to perform additional actions to minimize any additional releases to the environment as result of potential pool boiling. As such it is reasonable to conclude that any additional dose from pool boil-off is not significant, and it would be bounded by the conservatism in the FHA dose consequence analysis if reasonable assumptions are used.

Response to Item H

The minimum decay time for movement of irradiated fuel is contained in Section 3.9 of DCD, Chapter 16 (Technical Specifications), Revision 1:

> **TS 3.9.7, Refueling Operations – Decay Time**, LCO 3.9.7, states "The reactor shall be subcritical for at least 24 hours."

Response to Item I

The NRC code RADTRAD (v 3.0.3) was used to calculate the dose consequences due to a FHA. The following proprietary files are provided as an electronic attachment to this RAI response:

- Plant Scenario File: ESBWR FHA Rev1.psf
- Output File: ESBWR FHA Rev1.o0
- Release Fractions File: ESBWR FHA.rft
- Nuclide Inventory File: ESBWR FHA Rev1.nif

Response to Item J

DCD Tier 2, Revision 1, Subsection 15.4.1.4.2 lists 6 items to be confirmed by the COL application. This Subsection will be revised as indicated in the attached markups. Each item in DCD Tier 2, Revision 1, Subsection 15.4.1.4.2 is discussed below.

- **Normal intake flow rate into the Control Room is 14.2 m^3/min.:** The 14.2 m^3/min quoted in Revision 1 is being revised to 12.0 m^3/min (200 l/s). This item will be verified in system startup testing via an ITAAC item.
- **Emergency recirculation flow rate into and out of the Control Room is 7.1 m^3/min.:** The revised FHA dose consequence analysis does not credit the emergency filter trains, so this item is no longer applicable and is being deleted from the DCD.
- **The atmospheric dispersion factor for the Control Building is 1.0E-03 s/m^3:** The assumed X/Q values for the control room were listed previously in Item (C). This will be verified via a COL Action Item.
- **The Control Building HVAC system control system lag time is 10 seconds:** The revised FHA dose consequence analysis does not credit the emergency filter trains, therefore the time delay to system initiation is no longer applicable. This item is being deleted from the DCD.
- **The Control Room Habitability Area volume is 2625 m^3:** The assumed control room volume was reduced to 2200 m^3 (78000 ft^3). This value will be confirmed via an ITAAC item.
- **Fuel handling operations do not occur until 24 hours after shutdown:** Technical Specification Surveillance Requirement 3.9.7 requires a decay time of at least 24 hours.

This item is no longer required to be a COL or ITAAC item, thus it is deleted in the revision to DCD Tier 2.

Response to Item K

Regulatory Guide 1.183, Appendix B, Section 4.1 states that for the Fuel Building "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." Similarly, Section 5.3 states "If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period." As such, the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

The release rate quoted in Item II.C of DCD, Revision 1, Table 15.4-2 was chosen to ensure that >99.9% of the activity released to the Reactor Building was released in the 2 hour period following the event.

$$A_{\text{ReactorBldg}}\left(t=2hrs\right)=A_{0,\text{ReactorBldg}}e^{-(350\%/_{hr})(2hr)}=0.0009A_{0,\text{ReactorBldg}}$$

$$A_{\text{Released to Env.}}\left(t=2hrs\right)=1-A_{\text{ReactorBldg}}=0.9991A_{0,\text{ReactorBldg}}$$

GE interpreted this release rate as meeting the requirements of Regulatory Guide 1.183. However, in the revised FHA analysis for DCD revision 3 GE also assumes a very high release rate (1.0E+08 %/day) beginning at 1.95 hours to ensure all activity is released prior to 2 hours.

Response to Item L

The values listed in Table 15.4-3 were in error because they did not account for the DF of 200 for iodines. The entire table was revised to reflect the releases from the revised FHA dose consequence analysis.

DCD Impact:

DCD Tier 2, Subsections 15.4.1, 15.4.1.2.1, 15.4.1.3.3, 15.4.1.4.1, 15.4.1.4.2, 15.4.11 and Tables 15.4-2, 15.4-3 and 15.4-4.will be revised to reflect the revised FHA dose consequence analysis and are shown on the attached markup.

15.3 ANALYSIS OF INFREQUENT EVENTS

Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A, and Infrequent Events. Section 15.0 describes the licensing basis for this categorization.

The assumptions of Subsection 15.2.0 and Tables 15.2-1, 2 and 3 are applied in the TRACG calculations in Subsection 15.3.1, 2, 3, 4, 5, 6, 13, 14 and 15.

15.3.1 Loss Of Feedwater Heating With Failure of Selected Control Rod Run-In

15.3.1.1 Identification of Causes

A feedwater (FW) heater can be lost in at least two ways:

- steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating. The reference steam and power conversion system shown in Section 10.1 meets this requirement. In fact, the FW temperature drop based on the reference heat balance shown in Section 10.1 is less than 39°C (70°F). Therefore, the analyzed FW temperature drop shown in Table 15.2-1 is conservative.

This event conservatively assumes the loss of FW heating shown in Table 15.2-1, which causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic provided to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Rod Control and Information System (RC&IS) to initiate the selected control rods run-in (SCRRI) function to automatically reduce the reactor power and avoid a scram. However, for this event, SCRRI is assumed to fail and reactor scram on high simulated thermal power is not credited due to uncertainties. Therefore a new steady state is reached

15.3.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-2 lists the sequence of events for Figure 15.3-1.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the effects of this event. However, credit was not taken for this scram to consider the possibility that, for a similar case with a somewhat lower loss of heating, the scram setpoint might not be reached, while the consequences would only be slightly less severe for this case than the event analyzed here.

15.3.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Reactor scram should be initiated during this event. However, as explained above, credit for STPT scram was not taken. The nuclear system pressure does not significantly change [< 0.02 MPa (3.0 psi)] during the event, and consequently, the RCPB is not threatened.

15.3.1.4 Barrier Performance

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.1.5 Radiological Consequences

A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering transition boiling.

The scenario considered for the fission product release paths to the environment consists of the fission products traveling down the main steam lines, eventually reaching the condenser, where they leak from the condenser to the environment. This scenario is modeled after the BWR rod drop accident described in Regulatory Guide 1.183, Appendix C.

The source term for the event is defined in Tables 15.3-13, 15.3-14, and 15.3-15.

As can be seen in Table 15.3-16, the off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

15.3.2 Feedwater Controller Failure – Maximum Demand

15.3.2.1 Identification of Causes

See Subsection 15.2.4.2. This event assumes multiple control system failures, to simultaneously increase the flow in multiple FW pumps to their maximum limit.

15.3.2.2 Sequence of Events and Systems Operation

Sequence of Events

With excess FW flow, the water level rises to the high water level reference point (Level 8), at which time the FW pumps are run back, the main turbine is tripped and a scram is initiated. Table 15.3-3 lists the sequence of events for Figure 15.3-2. The figure shows the changes in important variables during this event.

Because Level 8 is located near the top of the separators, some moisture entrainment and carry-over to the turbine and bypass valve may occur. While this is potentially harmful to the turbine's integrity, it has no safety implications for the plant.

Identification of Operator Actions

The operator should:

- Follow the scram procedure.
- Observe that FW flow runback due to high water level has terminated the failure event.
- Switch the FW controller from auto to manual control to try to regain a correct output signal.
- Identify causes of the failure and report all key plant parameters during the event.

15.3.2.2.1 System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are tripping of the main turbine, FW flow runback, and scram due to high water level (Level 8).

15.3.2.3 Core and System Performance

15.3.2.3.1 Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 15.2-1.

Table 15.3-13

1000 Fuel Rod Failure Parameters

I. Data and assumptions used to estimate source terms		
A.	Power level, MWt	4590
B.	Number of bundles in core	1132
C.	Core fission product inventory released to coolant	Table 15.3-14
D.	Equivalent full length fuel rods per bundle	87.33
E.	Fuel rods damaged	1000
F.	Radial peaking factor for failed rods	1.5
II. Data and assumptions used to estimate activity released		
A.	Iodine released from failed fuel rods	10%
	Noble gases released from failed fuel rods	10%
	Alkali metals released from failed fuel rods	12%
B.	Iodine released from reactor coolant	10%
	Noble gases released from reactor coolant	100%
	Alkali metals released from reactor coolant	1%
C.	Iodine released from condenser	10%
	Noble gases released from condenser	100%
	Alkali metals released from condenser	1%
	Fission product inventory released to environment	Table 15.3-15
III. Dispersion and Dose Data		

Table 15.3-13

1000 Fuel Rod Failure Parameters

A. Meteorology:	
EAB	2.00E-03 s/m^3
LPZ	
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:	
0 – 2 hours	1.20E-03 s/m^3
2 – 8 hours	9.8E-04 s/m^3
8 – 24 hours	3.90E-04 s/m^3
1 – 4 days	3.80E-04 s/m^3
4 – 30 days	3.20E-04 s/m^3
C. Dose evaluations	Table 15.3-16

Table 15.3-14

1000 Fuel Rod Failure Fission Product Activity Released to Coolant

Isotope	Activity Released to Primary Coolant (MBq)
Kr-85	8.47E+07
Kr-85m	1.88E+09
Kr-87	3.63E+09
Kr-88	5.11E+09
Rb-86	1.94E+07
I-131	6.82E+09
I-132	9.92E+09
I-133	1.40E+10
I-134	1.55E+10
I-135	1.32E+10
Xe-133	1.40E+10
Xe-135	4.63E+09
Cs-134	1.63E+09
Cs-136	5.69E+08
Cs-137	1.06E+09

Table 15.3-15

1000 Fuel Rod Failure Fission Product Activity

Cumulative Release to Environment

	Cumulative Release to Environment (MBq)		
Isotope	**2 hours**	**8 hours**	**24 hours**
Kr-85	7.07E+04	2.82E+05	8.44E+05
Kr-85m	1.34E+06	3.56E+06	4.77E+06
Kr-87	1.79E+06	2.65E+06	2.69E+06
Kr-88	3.33E+06	7.40E+06	8.49E+06
Rb-86	1.61E+00	6.42E+00	1.89E+01
I-131	5.67E+04	2.24E+05	6.49E+05
I-132	6.12E+04	1.23E+05	1.34E+05
I-133	1.13E+05	4.09E+05	9.42E+05
I-134	6.24E+04	7.84E+04	7.86E+04
I-135	9.85E+04	2.95E+05	4.63E+05
Xe-133	1.16E+07	4.54E+07	1.30E+08
Xe-135	3.57E+06	1.15E+07	2.05E+07
Cs-134	1.36E+02	5.44E+02	1.63E+03
Cs-136	4.73E+01	1.88E+02	5.50E+02
Cs-137	8.84E+01	3.53E+02	1.06E+03

Table 15.3-16

1000 Fuel Rod Failure Dose Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	1.56E-01	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage	5.94E-02	2.5
Control Room Operator Dose for the Entire Period of the Accident	1.60E-01	5.0

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. 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 effected 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

$$E_f = \sigma_y \times \varepsilon \times Vol$$

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 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 energies 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 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 emergency 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. However, for the purposes of conservative estimation, no credit is taken for isolation or use of the HVAC System, and the gases are directly released to the environment at the rate identified in Table 15.4-3. Also, the Control Room ventilation is assumed to operate in normal mode. No credit is taken for mitigation from the charcoal. The total activity released to the environment is presented in Table 15.4-3.

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

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) 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

15.4.9.5.5 Results

The calculated exposures for the analysis are presented in Table 15.4-23 and are less than the regulatory guideline exposures.

15.4.10 Spent Fuel Cask Drop Accident

15.4.10.1 Identification of Causes

The fuel building design is such that a spent fuel cask drop height of 9.2m, 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.

15.4.10.2 Radiological Analysis

As stated above, the radiological consequences of this accident are not evaluated.

15.4.11 COL Information

COL applicants must confirm atmospheric dispersion factors for the following release locations:

- All release points must have an EAB X/Q value of less than or equal to that presented in Table 2.0-1 for all events.
- All release must have a LPZ X/Q values of less than or equal to those presented in Table 2.0-1 for all events.
- Fuel Handling Accident:
    - o Releases from the Reactor Building or the Fuel Building must have control room air intake X/Q values less than or equal to those presented in Table 2.0-1.
- Loss of Coolant Accident:
    - Releases from the Reactor Building, PCCS Ventilation Stack, and main Condenser must have control room louver X/Q values less than or equal to those presented in Table 2.0-1.
- Main steamline break analysis assumptions (Subsections 15.4.5.5.1 and 15.4.4.5.3)
- Feedwater line break analysis assumptions (Subsection 15.4.7.5.4)
- RWCU/SDC line break analysis assumptions (Subsection 15.4.9.5.4)

15.4.12 References

15.4-1 General Electric Co., "Radiological Accident Evaluation - The CONAC04A Code," NEDO-32708, August 1997.

15.4-2 Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document," Volume III.

15.4-3 General Electric Company, "Anticipated Chemical Behavior of Iodine under LOCA Conditions," NEDO-25370, January 1981.

Table 15.4-2

FHA Parameters

I. Data and Assumptions Used to Estimate Source Terms	
A. Power level, MWt	4590
B. 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. Radial peaking factor for damaged rods	1.5
D. Duration of accident, hr	2
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	≥7.01
C. Pool Retention decontamination factor	
Iodine (effective)	200
Noble gas	1
Alkali metals/particulates	Infinite

Table 15.4-2

FHA Parameters

C. Reactor Building release rate, %/hr	
0 – 1.95 hours	350
1.95 – 2.0 hours	1.0E+08
III. Control Room Parameters	
A. Control Room Volume, m^3	2.2E+03
B. Unfiltered intake, l/s	200
C. Filtered intake, l/s	0
D. Unfiltered inleakage, l/s	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, sec./m^3	2.00E-03
Control Room	
Reactor Building Release	Table 2.0-1
Fuel Building Release	Table 2.0-1
B. Dose conversion assumptions	RG 1.183
C. Activity inventory/releases	Table 15.4-3
D. Dose evaluations	Table 15.4-4

Table 15.4-3

FHA Isotopic Release to Environment

Isotope	Activity (MBq)
I-131	**4.4E+06**
I-132	**2.9E+03**
I-133	**2.8E+06**
I-134	**3.2E-02**
I-135	**4.6E+05**
Kr-85m	**7.8E+06**
Kr-85	**1.5E+07**
Kr-87	**5.8E+02**
Kr-88	**1.2E+06**
Xe-133	**1.1E+09**
Xe-135	**6.6E+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:		
Exclusion Area Boundary (EAB) for a 2-hour duration	3.6	6.3
Outer boundary of Low Population Zone (LPZ) for a 2-hour duration	3.6	6.3
Control Room dose for the duration of the accident	2.3	5.0

ENCLOSURE 2

MFN 07-017

Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis RAI Number 15.3-25 and 15.4-1

Contains GE Proprietary Information

PROPRIETARY INFORMATION NOTICE

CONTENTS OF CD:

ESBWR FHA Rev1.nif
ESBWR FHA Rev1.o0
ESBWR FHA Rev1.psf
ESBWR FHA.rft
ESBWR LOFWH_Condenser – No EBAS Rev. 2o0
ESBWR LOFWH_Condenser – No EBAS Rev2.psf
ESBWR LOFWH_Condenser.nif
ESBWR LOFWH_Condenser.rft

Enclosure 3

MFN 07-017

David Hinds

Affidavit

General Electric Company

AFFIDAVIT

I, David Hinds, state as follows:

(1) I am Manager, New Units Engineering, General Electric Company ("GE"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 2 of GE's letter, MFN 07-017, James Kinsey to NRC, entitled *Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis RAI Numbers 15.3-24, 15.3-25, and 15.4-1*and dated February 16, 2007. The proprietary information in the Enclosure 2 consists of RADTRAD input and output files that are provided in electronic form on a CD labeled as GE proprietary information.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GE's competitors without license from GE constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GE customer-funded development plans and programs, resulting in potential products to GE;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed ESBWR design information developed by GE and/or its partners over a period of more than ten years at a cost of several million dollars. This information, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

The development of the provided information along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 16th day of February, 2007

David Hinds
General Electric Company