HITACHI

GE Hitachi Nuclear Energy

James C. Kinsey
Vice President, 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 06-396, Supplement 2
MFN 07-100, Supplement 2 and
MFN 07-197, Supplement 1

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Subject: **Response to Portion of NRC Request for Additional Information Letter Nos. 69 and 109 Related to ESBWR Design Certification Application – X/Q Values Used in Radiological Dose Analyses – RAI Numbers 2.3-8S01, 15.4-3S01 and 15.4-4S01**

The purpose of this letter is to submit the GE-Hitachi Nuclear Energy (GEH) response to a portion of the U.S. Nuclear Regulatory Commission Requests for Additional Information (RAI) sent by Reference 1 for RAI 2.3-8, Supplement 1, and Reference 2 for RAIs 15.4-3, Supplement 1 and 15.4-4, Supplement 1. Reference 3 provided RAI No. 2.3-8 for which the response was provided in Reference 4. Reference 5 provided RAI Nos. 15.4-3 and 15.4-4 for which responses were provided in References 6 and 7, respectively. Responses to RAI Nos. 2.3-8, Supplement 1, 15.4-3, Supplement 1, and 15.4-4 Supplement 1 are provided in Enclosure 1. DCD markup pages required by these responses are provided in Enclosure 2.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

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References:

1. MFN 07-555, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request for Additional Information Letter No. 109 Related to ESBWR Design Certification Application,* October 12, 2007.
2. E-mail from U.S. Nuclear Regulatory Commission, June 13, 2007.
3. MFN 06-201, Letter from U.S. Nuclear Regulatory Commission to David H. Hinds, *Request for Additional Information Letter No. 37 Related to ESBWR Design Certification Application,* June 21, 2006.
4. MFN 06-396, Letter from GEH to U.S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 37 Related to ESBWR Design Certification Application – Siting Issues – RAI Nos. 2.1-2, 2.3-7, 2.3-8, 2.3-10, 14.3-23, 14.3-24, 14.3-25, and 15.3-2,* October 20, 2006.
5. MFN 06-381, Letter from U.S. Nuclear Regulatory Commission to David H. Hinds, *Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application,* October 11, 2006.
6. MFN 07-100, Letter from GEH to U.S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application – Safety Analysis – RAI Nos. 15.4-2 and 15.4-3,* March 26, 2007.
7. MFN 07-197, Letter from GEH to U.S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application – Safety Analysis – RAI No. 15.4-4,* April 12, 2007.

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter Nos 69 and 109 Related to ESBWR Design Certification Application – Safety Analyses – RAI Numbers 2.3-8S01, 15.4-3S01 and 15.4-4S01
2. DCD Markups

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	eDRF	0000-0077-2899

Enclosure 1

MFN 06-396, Supplement 2,
MFN 07-100, Supplement 2 and
MFN 07-197, Supplement 1

Response to Portion of NRC Request for Additional Information Letter Nos. 69 and 109 Related to ESBWR Design Certification Application

X/Q Values Used in Radiological Dose Analyses

RAI Numbers 2.3-8 S01, 15.4-3 S01 & 15.4-4 S01

NRC RAI 2.3-8 S01:

DCD Tier 2 Tables 15.4-14 and 15-4.21 indicate that a X/Q value of 1.00 × 10^{-3} s/m3 is used to calculate doses at the EAB for the feedwater line break and the RWCU/SDC line break accidents, respectively. Please explain why the EAB X/Q value used in these radiological consequence analyses differs from the EAB X/Q value of 2.00 × 10^{-3} s/m3 listed as a standard plant site design parameter in DCD Tier 1 Table 5.1-1 and Tier 2 Table 2.0-1. The use of a lower EAB X/Q value in these DCD radiological consequence analyses results in lower calculated doses for the EAB.

GEH Response:

The RWCU/SDC line break accident analysis and DCD Table 15.4-21 were revised in response to NRC RAI 15.4-4 (MFN 07-197, April, 11 2007). An EAB X/Q value of $2.0x10^{-3}$ s/m^3 was used in the revised analysis and is reflected in DCD Tier 2 Revision 4, Table 15.4-21.

The Feedwater Line Break accident has been revised in response to this RAI. An EAB X/Q value of $2.0x10^{-3}$ s/m^3 is used in the revised analysis and is reflected in the attached markup of DCD Table 15.4-14. Other revised feedwater line break accident analysis descriptions, assumptions and results are reported in the attached markups of DCD Section 15.4.7 and Tables 15.4-15 and 15.4-16.

DCD Impact:

DCD Tier 2, Section 15.4.7 and Tables 15.4-14, 15.4-15, and 15.4-16 will be revised as noted on the attached markups in DCD Tier 2, Revision 5.

NRC RAI 15.4-3 S01:

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006 GE response in MFN 07-100 dated March 26, 2007

(1) Add the following information to Tables 15.4-14 and 15.4-17: Duration of accident, EAB, LPZ, and control room X/Q values, Release point, Control room operator doses, Control room not isolated, Control room normal ventilation system will be in operation during this event

(2) Revise the following information in Tables 15.4-19 and 23:

a) Reword EAB to read ... for any (worst) 2 hours rather than for the entire period of the radioactive cloud passage.

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

GEH Response:

(1) The duration of the accident for EAB, LPZ and control room X/Q values, release point, control room operator doses, that the control room is not isolated during the event, and the control room ventilation system is in normal operation, is provided in the attached markups of DCD Tables 15.4-14 and 15.4-17. Additional changes to Tables 15.4-14, 15.4-17 and 15.4-18 are included in the attached markups to provide analysis parameter values in both SI and US Customary units where appropriate.

(2) DCD Tables 15.4-19 and 23 will be revised to clarify that the EAB cloud passage is considered for the worst 2 hours, the LPZ and control room doses are considered for 0 to 30 day accident duration.

DCD Impact:

DCD Tier 2, Tables 15.4-14, 15.4-17, 15.4-18, 15.4-19 and 15.4-23 will be revised as noted on the attached markups in DCD Tier 2, Revision 5.

NRC RAI 15.4-4 S01:

Reference: RAI 15.3-25 in NRC letter dated October 11, 2006 GE response in MFN 07-197 dated April 12, 2007

(1) Add the following information to Table 15.4-21: Duration of accident EAB, LPZ, and control room X/Q values, Release point, Control room operator doses, Control room not isolated, Control room normal ventilation system will be in operation during this event

(2) The revisions of Table 15.4-19 and 23 as indicated in Supplement No. 1 RAI 15.4-3 above are also required for resolution of this RAI.

GEH Response:

(1) The duration of the accident for EAB, LPZ and control room X/Q values, release point, control room operator doses, that the control room is not isolated during the event, and the control room ventilation system is in normal operation will be added to Table 15.4-21. See attached DCD markups.

(2) Tables 15.4-19 and 15.4-23 will be revised to clarify that the EAB cloud passage is considered for the worst 2 hours, and the LPZ and control room doses are considered for 0 to 30 day accident duration. Additional changes to Tables 15.4-21 and 15.4-22 are included in the attached markups to provide analysis parameter values in both SI and US Customary units where appropriate.

Other changes were made to the Instrument Line Break and Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) Line Break accident analyses in response to this RAI. These analyses were revised to provide consistency between related DCD Section 15.4 accident analysis reactor coolant source term assumptions. Revised Small Line Break accident analysis descriptions, assumptions and results are reported on the attached markups of DCD Section 15.4.8, and Tables 15.4-17, 15.4-18 and 15.4-19. Revised Reactor Water Cleanup/Shutdown Cooling System line break accident analysis descriptions, assumptions and results are reported on the attached markups of DCD Section 15.4.9 and Tables 15.4-22 and 15.4-23.

DCD Impact:

DCD Tier 2, Sections 15.4-8 and 15.4-9, and Tables 15.4-17, 15.4-18, 15.4-19, 15.4-21, 15.4-22, and 15.4-23 will be revised as noted on the attached markups in DCD Tier 2, Revision 5.

Enclosure 2

MFN 06-396, Supplement 2,
MFN 07-100, Supplement 2 and
MFN 07-197, Supplement 1

DCD Markups

15.4.7 Feedwater Line Break Outside Containment

The feedwater line break containment response evaluation is provided in Section 6.2

The feedwater line break ECCS capability evaluation is provided in Section 6.3.

The feedwater line break radiological evaluation is as follows:

The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation for this type of break. ~~The break is assumed to be~~An instantaneous, circumferential break of the feedwater line at a location ~~and~~ downstream of the high pressure feedwater heaters and upstream of the outermost containment isolation valve is conservatively assumed. This location corresponds to the highest temperature condition for the Feedwater System and is selected to maximize the fraction of radionuclides in liquid feedwater that become airborne as a result of a feedwater line break.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break Inside Containment) has been quantitatively analyzed and is presented in Section 6.3. Therefore, the following discussion provides only new information that is not presented in Section 6.3. All other information is cross-referenced to appropriate Chapter 6 subsections.

15.4.7.1 Identification of Causes

A feedwater line break is assumed without ~~the~~ an identified cause~~ being identified~~. The subject piping is designed to high quality~~, to~~ engineering codes and standards, and to seismic environmental requirements.

15.4.7.2 Sequence of Events and System Operation

15.4.7.2.1 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator action is required or credited in the radiological consequence analysis.

However, the operator should perform the following (shown for informational purposes only) actions:

- Determine that a line break has occurred and evacuate the area of the Turbine Building.
- Ensure that the reactor is shut down and that the ICS and the CRD systems are operating normally or, if failed, that the ADS and GDCS are operating.
- Implement site radiation incident procedures.
- Shut down the feedwater system and de-energize any electrical equipment that may be damaged by water from the feedwater system in the Turbine Building.
- Continue to monitor reactor water level and the performance of the ECCS while the radiation incident procedure is being implemented and begin normal reactor cooldown measures.
- Initiate the FAPCS in the suppression pool cooling mode (if necessary) and RWCU/SDC in the shutdown cooling mode.

These actions occur over an elapsed time of 3-4 hours.

15.4.7.2.2 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the ECCS. The Reactor Protection System, SRVs, ECCS, and Control Rod Drive system are assumed to function properly to ensure a safe shutdown.

The ESF systems, including the ADS and GDCS, are assumed to operate normally.

15.4.7.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the general LOCA break spectrum presented in detail within Section 6.3. The general single-failure analysis for LOCAs is presented in detail in Subsection 6.3.3. For the feedwater line break outside the containment, the worst single failure does not result in core uncovery, and there ~~would be~~is no fuel damage.

15.4.7.3 Core and System Performance

15.4.7.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative and ~~envelope~~ envelop~~ing~~es ~~assessment of~~ the consequences of the postulated failure (i.e., severance) of ~~aone the~~ feedwater pipe~~ing~~ line~~s~~ external to the containment.

15.4.7.3.2 Qualitative Results

The feedwater line break outside containment is less limiting, from a core performance evaluation standpoint, than the main steamline break outside the containment analysis presented in Subsection 15.4.5 and the LOCA inside the containment analysis presented in ~~and~~ 15.4.4.

The break is isolated by closure of the feedwater check valves. The main steamlines are isolated on water level 2, and the ADS and the GDCS together restore the reactor water level to the normal elevation. The fuel is covered throughout the transient and there is no pressure or temperature transient sufficient to cause fuel damage.

15.4.7.3.3 Consideration of Uncertainties

This event is conservatively analyzed and uncertainties ~~were~~ adequately considered (see Section 6.3).

15.4.7.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in piping connected to the reactor coolant pressure boundary or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.4.5. The feedwater system piping break is less severe than the main steamline break. Results of analysis of this event can be found in Section 6.3.

15.4.7.5 Radiological Consequences

15.4.7.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC. ~~However, for consistency, the RG 1.183 guideline exposure acceptance criteria for the MSLBA are used for the Feedwater Line Break accident.~~

~~The analysis is based on a conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are presented in Reference 15.4-1.~~ Specific values of parameters used in the evaluation are presented in Table 15.4-14.

~~15.4.7.5.2 Fission Product Release~~

Source Term: There is no fuel damage as a consequence of this accident. ~~In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to the occurrence of the break) is released from the contained piping system prior to isolation closure.~~The only radionuclide source available for release from a feedwater line break is that contained in the Condensate and Feedwater System prior to the break, and extraction steam supplied to feedwater heaters over the assumed event duration.

The release duration is conservatively analyzed based on the amount of time required to empty the condenser hotwell, assuming condensate system design flow rate. The release duration is calculated to be 230 seconds based on a hotwell storage capacity of 400 m^3 and a condensate design flow rate of 1743 kg/s. The mass of feedwater released from the break is calculated to be 5.6E+05 kg based on the 230-second release duration and a total feedwater design flow rate of 2427 kg/s.

~~The iodine concentration assumed is that of the maximum equilibrium reactor water concentration used for the MSLBA~~Reactor coolant radionuclide source terms are calculated consistent with iodine spiking assumptions provided in Regulatory Guide 1.183 ~~iodine spiking assumptions provided~~ for analyzing consequences of a Main Steam Line Break for a BWR, ~~as~~which is presented in Subsection 15.4.5.5.1. ~~, subject to a 2%~~Separate carryover fractions for ~~of~~ iodine and non-iodine particulate species present in ~~the~~ reactor coolant are then applied ~~water~~ to obtain steam and condensate radionuclide source terms consistent with the assumptions presented in Table 11.1-3. No credit for radionuclide removal by the Condensate Purification System is assumed over the duration of the release. Noble gas activity in the condensate is negligible and is ~~therefore~~ ignored in this analysis~~, because the air ejectors remove all noble gases from the condenser~~.

~~15.4.7.5.3 Fission Product Transport to the Environment~~

Fission Product Transport to the Environment: The transport pathway consists of liquid release from the break, carryover to the Turbine Building atmosphere due to flashing and partitioning, and unfiltered release to the environment through the Turbine Building ~~ventilation system~~walls.

~~Taking no~~No credit is taken for holdup, decay or plate-out during transport through the Turbine Building. ~~, the~~ The activity released of ~~activity~~ to the environment is presented in Table 15.4-15.

Control Room: Control Room ventilation is assumed to operate in the normal mode for the duration of the event. No credit is taken for Control Room isolation, or operation of the Control Room emergency filter units (EFU).

~~15.4.7.5.4 Assumptions to be Confirmed by the COL Applicant~~

Assumptions Requiring Confirmation: Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.~~The following are assumptions in the radiological analysis that require confirmation:~~

- ~~The main condenser is sized for at least 2 minutes worth of main steam flow.~~
- ~~The demineralizer efficiency is at least 99% (all coolant that is released during the accident is filtered through the demineralizer).~~

~~15.4.7.5.5~~ 15.4.7.5.2 Results

The calculated exposures for the analysis are presented in Table 15.4-16, and are less than the regulatory guideline exposures.

~~Main Steam Line Break Accident~~

Table 15.4-14

Feedwater Line Break Accident Parameters

I. Data and Assumptions Used to Estimate Source Terms	
A. Fuel Damage	None
B. Reactor Coolant Activity, Bq/g (μCi/g) DE I-131 Pre-incident Spike Equilibrium Iodine Activity	 148,000 (4.0) 7400 (0.2)
C. Water Mass Released, kg (lbm)	5.6E+05 (1.2E+06)
II. Data and Assumptions Used to Estimate Activity Released	
A. Water-to-Steam Flashing Fractions	0.232
B. Iodine Plateout Fraction, %	0
C. Release Duration	230 seconds
D. Release Point	Turbine Building
E. Turbine Building Flow rate to Environment, %/hour	Instantaneous
III. Control Room Parameters	
A. Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E4)
B. Unfiltered intake, l/s (ft^3/min)	200 (424)
C. Filtered intake, l/s (ft^3/min)	0 (0)
D. Unfiltered inleakage, l/s (ft^3/min)	0 (0)
E. Occupancy Factors	
0 – 1 day	1.0
1 – 4 days	0.6
4 – 30 days	0.4
IV. Dispersion and Dose Data	
A. Offsite Meteorology Exclusion Area Boundary 0-2 Hours Low Population 0-8 Hours	 2.0E-03 s/m^3 1.9E-04 s/m^3

Table 15.4-14

Feedwater Line Break Accident Parameters

> 8 Hours	NR*
B. Control Room Meteorology (Turbine Building Release Point)	
0-2 Hours	1.2E-03 s/m^3
> 2 Hours	NR*
C. Method of Dose Calculation	RG 1.183
D. Activity Inventory/Releases	Table 15.4-15
E. Dose Results	Table 15.4-16

* Due to the short release, values > 2 hours do not impact the calculated doses~~,~~; therefore, they are Not Required (NR).

~~**Table 15.4-15**~~

~~**Feedwater Line Break Accident**~~

~~**Environment Releases**~~

~~Isotope~~	~~Activity (MBq)~~
~~I-131~~	~~1.3E+02~~
~~I-132~~	~~1.2E+03~~
~~I-133~~	~~8.7E+02~~
~~I-134~~	~~2.2E+03~~
~~I-135~~	~~1.2E+03~~

Table 15.4-15

Feedwater Line Break Accident Isotopic Release to Environment

	Equilibrium Iodine		Pre-incident Spike	
Isotope	MBq	Ci	MBq	Ci
I-131	7.64E+03	2.06E-01	1.53E+05	4.13E+00
I-132	7.28E+04	1.97E+00	1.46E+06	3.93E+01
I-133	5.09E+04	1.38E+00	1.02E+06	2.75E+01
I-134	1.35E+05	3.64E+00	2.69E+06	7.28E+01
I-135	7.64E+04	2.06E+00	1.53E+06	4.13E+01
Cs-134	2.65E+00	7.16E-05	2.65E+00	7.16E-05
Cs-136	1.77E+00	4.77E-05	1.77E+00	4.77E-05
Cs-137	7.06E+00	1.91E-04	7.06E+00	1.91E-04
Co-58	2.16E+00	5.85E-05	2.16E+00	5.85E-05
Co-60	4.28E+00	1.16E-04	4.28E+00	1.16E-04
Sr-89	9.93E+00	2.68E-04	9.93E+00	2.68E-04
Sr-90	6.84E-01	1.85E-05	6.84E-01	1.85E-05
Y-90	6.84E-01	1.85E-05	6.84E-01	1.85E-05
Sr-91	3.75E+02	1.01E-02	3.75E+02	1.01E-02
Sr-92	9.05E+02	2.45E-02	9.05E+02	2.45E-02
Y-91	3.97E+00	1.07E-04	3.97E+00	1.07E-04
Y-92	5.52E+02	1.49E-02	5.52E+02	1.49E-02

Table 15.4-15

Feedwater Line Break Accident Isotopic Release to Environment

	Equilibrium Iodine		Pre-incident Spike	
Isotope	MBq	Ci	MBq	Ci
Y-93	3.75E+02	1.01E-02	3.75E+02	1.01E-02
Zr-95	7.95E-01	2.15E-05	7.95E-01	2.15E-05
Nb-95	7.95E-01	2.15E-05	7.95E-01	2.15E-05
Mo-99	1.96E+02	5.31E-03	1.96E+02	5.31E-03
Tc-99m	1.96E+02	5.31E-03	1.96E+02	5.31E-03
Ru-103	1.96E+00	5.31E-05	1.96E+00	5.31E-05
Ru-106	2.87E-01	7.76E-06	2.87E-01	7.76E-06
Te-129m	3.97E+00	1.07E-04	3.97E+00	1.07E-04
Te-131m	9.71E+00	2.62E-04	9.71E+00	2.62E-04
Te-132	9.93E-01	2.68E-05	9.93E-01	2.68E-05
Ba-140	3.97E+01	1.07E-03	3.97E+01	1.07E-03
La-140	3.97E+01	1.07E-03	3.97E+01	1.07E-03
Ce141	2.87E+00	7.76E-05	2.87E+00	7.76E-05
Ce-144	2.87E-01	7.76E-06	2.87E-01	7.76E-06
Np-239	7.95E+02	2.15E-02	7.95E+02	2.15E-02

Table 15.4-16

Feedwater Line Break Analysis 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~~	~~1.7E-04~~	~~2.5~~
~~Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage~~	~~1.7E-04~~	~~2.5~~

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE,** Sv (REM)	**Acceptance Criterion TEDE,** Sv
Exclusion Area Boundary (EAB) for any (Worst) 2-hour Period		
Equilibrium Iodine Spike Case	2.0E-04 (0.020)	0.025
Pre-incident Iodine Spike Case	0.0039 (0.39)	0.25
Outer Boundary of Low Population Zone (LPZ) for Duration of the Accident (30 days)		
Equilibrium Iodine Spike Case	1.9E-05 (1.9E-03)	0.025
Pre-incident Iodine Spike Case	3.8E-04 (0.038)	0.25
Control Room Dose for the Duration of the Accident (30 days)		
Equilibrium Iodine Case	7.1E-05 (7.1E-03)	0.05
Pre-incident Iodine Spike Case	0.0014 (0.14)	0.05

~~15.0.1.1.1~~ 15.4.8.5.2 Fission Product Release

Two cases exist for the iodine coolant concentration: one for the maximum equilibrium iodine and one for the pre-accident "iodine spike." It is conservatively assumed that the release to the environment is instantaneous, with no iodine plateout. ~~The iodine activity in the coolant is assumed to be at the maximum equilibrium Technical Specification limit (see MSLBA in Subsection 15.4.5.5) for continuous operation.~~ Based on data in Table 15.4-17, the integral radioisotope release to the environment ~~amount of iodine released to the Reactor Building atmosphere and to the environment~~ is presented in Tables 15.4-18a and 15.4-18b.

Table 15.4-17
Instrument Line Break Accident Parameters

I. Data and assumptions used to estimate source terms	
Fuel Damage	none
A. Reactor Coolant Iodine Activity	
Equilibrium Concentration, Bq/g (μCi/g) DE I-131	7,400 (0.2)
Pre-Accident Spike Concentration, Bq/g (μCi/g) DE I-131	148,000 (4.0)
B. Mass of fluid released, kg (lbm)	14,785 (32,565)
C. Mass of fluid flashed to steam, kg (lbm)	4,007 (8,825)
D. Duration of release, hr	6
II. Data and assumptions used to estimate activity released	
A. Iodine plateout fraction, %	0
B. Reactor Building Release rate, %/hour	5.00E+09
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. Occupancy Factors	
0 – 1 days	1.0
1 – 4 days	0.6
4 – 30 days	0.4
IV. Reactor Building Parameters	
A. Reactor Building Volume, m^3 (ft^3)	2.4E4 (8.5E+05)
B. Reactor Building Leakage rate, %/hour	5.00E+09
IV. Dispersion and Dose Data (X/Q)	
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	

Table 15.4-17
Instrument Line Break Accident Parameters

0 – 2 hours	1.50E-03 s/m^3
2 – 8 hours	1.10E-03 s/m^3
8 – 24 hours	5.00E-04 s/m^3
1 – 4 days	4.20E-04 s/m^3
4 – 30 days	3.80E-04 s/m^3
B. Method of Dose Calculation	RG 1.183
C. Dose evaluations	Table 15.4-19

Table 15.4-18a
ILB Accident Integral Release to the Environment for the Pre-Accident Spike Case

Time (hr)	0.5		1		2		4		6	
	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci
Co-58	1.5E+01	4.0E-04	2.3E+01	6.3E-04	4.1E+01	1.1E-03	6.5E+01	1.7E-03	6.7E+01	1.8E-03
Co-60	2.9E+01	7.9E-04	4.6E+01	1.3E-03	8.1E+01	2.2E-03	1.3E+02	3.5E-03	1.3E+02	3.6E-03
Sr-89	6.7E+01	1.8E-03	1.1E+02	2.9E-03	1.9E+02	5.1E-03	3.0E+02	8.0E-03	3.1E+02	8.3E-03
Sr-90	4.6E+00	1.3E-04	7.4E+00	2.0E-04	1.3E+01	3.5E-04	2.0E+01	5.5E-04	2.1E+01	5.7E-04
Sr-91	2.5E+03	6.9E-02	4.1E+03	1.1E-01	7.1E+03	1.9E-01	1.1E+04	3.0E-01	1.2E+04	3.1E-01
Sr-92	6.1E+03	1.7E-01	9.8E+03	2.7E-01	1.7E+04	4.6E-01	2.7E+04	7.3E-01	2.8E+04	7.5E-01
Y-90	4.6E+00	1.3E-04	7.4E+00	2.0E-04	1.3E+01	3.5E-04	2.0E+01	5.5E-04	2.1E+01	5.7E-04
Y-91	2.7E+00	7.3E-05	4.3E+00	1.2E-04	7.5E+00	2.0E-04	1.2E+01	3.2E-04	1.2E+01	3.3E-04
Y-92	3.7E+03	1.0E-01	6.0E+03	1.6E-01	1.0E+04	2.8E-01	1.7E+04	4.5E-01	1.7E+04	4.6E-01
Y-93	2.5E+02	6.9E-03	4.1E+02	1.1E-02	7.1E+02	1.9E-02	1.1E+03	3.0E-02	1.2E+03	3.1E-02
Zr-95	5.4E+00	1.5E-04	8.6E+00	2.3E-04	1.5E+01	4.1E-04	2.4E+01	6.4E-04	2.5E+01	6.6E-04
Nb-95	5.4E+00	1.5E-04	8.6E+00	2.3E-04	1.5E+01	4.1E-04	2.4E+01	6.4E-04	2.5E+01	6.6E-04
Mo-99	1.3E+03	3.6E-02	2.1E+03	5.8E-02	3.7E+03	1.0E-01	5.9E+03	1.6E-01	6.1E+03	1.6E-01
Tc-99m	1.3E+03	3.6E-02	2.1E+03	5.8E-02	3.7E+03	1.0E-01	5.9E+03	1.6E-01	6.1E+03	1.6E-01
Ru-103	1.3E+01	3.6E-04	2.1E+01	5.8E-04	3.7E+01	1.0E-03	5.9E+01	1.6E-03	6.1E+01	1.6E-03
Ru-106	1.9E+00	5.3E-05	3.1E+00	8.4E-05	5.4E+00	1.5E-04	8.6E+00	2.3E-04	8.8E+00	2.4E-04
Te-129m	2.7E+01	7.3E-04	4.3E+01	1.2E-03	7.5E+01	2.0E-03	1.2E+02	3.2E-03	1.2E+02	3.3E-03
Te-131m	6.6E+01	1.8E-03	1.1E+02	2.8E-03	1.8E+02	5.0E-03	2.9E+02	7.8E-03	3.0E+02	8.1E-03
Te-132	6.7E+00	1.8E-04	1.1E+01	2.9E-04	1.9E+01	5.1E-04	3.0E+01	8.0E-04	3.1E+01	8.3E-04
I-131	5.4E+04	1.5E+00	8.7E+04	2.3E+00	1.5E+05	4.1E+00	2.4E+05	6.5E+00	2.5E+05	6.7E+00
I-132	4.9E+05	1.3E+01	7.8E+05	2.1E+01	1.4E+06	3.7E+01	2.2E+06	5.8E+01	2.2E+06	6.0E+01
I-133	3.6E+05	9.6E+00	5.7E+05	1.5E+01	9.9E+05	2.7E+01	1.6E+06	4.2E+01	1.6E+06	4.4E+01
I-134	8.9E+05	2.4E+01	1.4E+06	3.8E+01	2.5E+06	6.7E+01	3.9E+06	1.1E+02	4.0E+06	1.1E+02
I-135	4.9E+05	1.3E+01	7.9E+05	2.1E+01	1.4E+06	3.7E+01	2.2E+06	5.9E+01	2.2E+06	6.1E+01
Cs-134	1.8E+01	4.9E-04	2.9E+01	7.8E-04	5.0E+01	1.4E-03	7.9E+01	2.1E-03	8.2E+01	2.2E-03
Cs-136	1.2E+01	3.2E-04	1.9E+01	5.2E-04	3.3E+01	9.1E-04	5.3E+01	1.4E-03	5.4E+01	1.5E-03
Cs-137	4.8E+01	1.3E-03	7.7E+01	2.1E-03	1.3E+02	3.6E-03	2.1E+02	5.7E-03	2.2E+02	5.9E-03
Ba-140	2.7E+02	7.3E-03	4.3E+02	1.2E-02	7.5E+02	2.0E-02	1.2E+03	3.2E-02	1.2E+03	3.3E-02
La-140	2.7E+02	7.3E-03	4.3E+02	1.2E-02	7.5E+02	2.0E-02	1.2E+03	3.2E-02	1.2E+03	3.3E-02
Ce-141	1.9E+01	5.3E-04	3.1E+01	8.4E-04	5.4E+01	1.5E-03	8.6E+01	2.3E-03	8.8E+01	2.4E-03
Ce-144	1.9E+00	5.3E-05	3.1E+00	8.4E-05	5.4E+00	1.5E-04	8.6E+00	2.3E-04	8.8E+00	2.4E-04
Np-239	5.4E+03	1.5E-01	8.6E+03	2.3E-01	1.5E+04	4.1E-01	2.4E+04	6.4E-01	2.5E+04	6.6E-01

Table 15.4-18b
ILB Accident Integral Release to the Environment for the Equilibrium Case

Time (hr)	0.5		1		2		4		6	
	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci
Co-58	1.5E+01	4.0E-04	2.3E+01	6.3E-04	4.1E+01	1.1E-03	6.5E+01	1.7E-03	6.7E+01	1.8E-03
Co-60	2.9E+01	7.9E-04	4.6E+01	1.3E-03	8.1E+01	2.2E-03	1.3E+02	3.5E-03	1.3E+02	3.6E-03
Sr-89	6.7E+01	1.8E-03	1.1E+02	2.9E-03	1.9E+02	5.1E-03	3.0E+02	8.0E-03	3.1E+02	8.3E-03
Sr-90	4.6E+00	1.3E-04	7.4E+00	2.0E-04	1.3E+01	3.5E-04	2.0E+01	5.5E-04	2.1E+01	5.7E-04
Sr-91	2.5E+03	6.9E-02	4.1E+03	1.1E-01	7.1E+03	1.9E-01	1.1E+04	3.0E-01	1.2E+04	3.1E-01
Sr-92	6.1E+03	1.7E-01	9.8E+03	2.7E-01	1.7E+04	4.6E-01	2.7E+04	7.3E-01	2.8E+04	7.5E-01
Y-90	4.6E+00	1.3E-04	7.4E+00	2.0E-04	1.3E+01	3.5E-04	2.0E+01	5.5E-04	2.1E+01	5.7E-04
Y-91	2.7E+00	7.3E-05	4.3E+00	1.2E-04	7.5E+00	2.0E-04	1.2E+01	3.2E-04	1.2E+01	3.3E-04
Y-92	3.7E+03	1.0E-01	6.0E+03	1.6E-01	1.0E+04	2.8E-01	1.7E+04	4.5E-01	1.7E+04	4.6E-01
Y-93	2.5E+02	6.9E-03	4.1E+02	1.1E-02	7.1E+02	1.9E-02	1.1E+03	3.0E-02	1.2E+03	3.1E-02
Zr-95	5.4E+00	1.5E-04	8.6E+00	2.3E-04	1.5E+01	4.1E-04	2.4E+01	6.4E-04	2.5E+01	6.6E-04
Nb-95	5.4E+00	1.5E-04	8.6E+00	2.3E-04	1.5E+01	4.1E-04	2.4E+01	6.4E-04	2.5E+01	6.6E-04
Mo-99	1.3E+03	3.6E-02	2.1E+03	5.8E-02	3.7E+03	1.0E-01	5.9E+03	1.6E-01	6.1E+03	1.6E-01
Tc-99m	1.3E+03	3.6E-02	2.1E+03	5.8E-02	3.7E+03	1.0E-01	5.9E+03	1.6E-01	6.1E+03	1.6E-01
Ru-103	1.3E+01	3.6E-04	2.1E+01	5.8E-04	3.7E+01	1.0E-03	5.9E+01	1.6E-03	6.1E+01	1.6E-03
Ru-106	1.9E+00	5.3E-05	3.1E+00	8.4E-05	5.4E+00	1.5E-04	8.6E+00	2.3E-04	8.8E+00	2.4E-04
Te-129m	2.7E+01	7.3E-04	4.3E+01	1.2E-03	7.5E+01	2.0E-03	1.2E+02	3.2E-03	1.2E+02	3.3E-03
Te-131m	6.6E+01	1.8E-03	1.1E+02	2.8E-03	1.8E+02	5.0E-03	2.9E+02	7.8E-03	3.0E+02	8.1E-03
Te-132	6.7E+00	1.8E-04	1.1E+01	2.9E-04	1.9E+01	5.1E-04	3.0E+01	8.0E-04	3.1E+01	8.3E-04
I-131	2.7E+03	7.3E-02	4.3E+03	1.2E-01	7.6E+03	2.1E-01	1.2E+04	3.2E-01	1.2E+04	3.3E-01
I-132	2.4E+04	6.6E-01	3.9E+04	1.1E+00	6.8E+04	1.8E+00	1.1E+05	2.9E+00	1.1E+05	3.0E+00
I-133	1.8E+04	4.8E-01	2.8E+04	7.7E-01	5.0E+04	1.3E+00	7.8E+04	2.1E+00	8.1E+04	2.2E+00
I-134	4.4E+04	1.2E+00	7.1E+04	1.9E+00	1.2E+05	3.4E+00	2.0E+05	5.3E+00	2.0E+05	5.5E+00
I-135	2.5E+04	6.7E-01	3.9E+04	1.1E+00	6.9E+04	1.9E+00	1.1E+05	2.9E+00	1.1E+05	3.0E+00
Cs-134	1.8E+01	4.9E-04	2.9E+01	7.8E-04	5.0E+01	1.4E-03	7.9E+01	2.1E-03	8.2E+01	2.2E-03
Cs-136	1.2E+01	3.2E-04	1.9E+01	5.2E-04	3.3E+01	9.1E-04	5.3E+01	1.4E-03	5.4E+01	1.5E-03
Cs-137	4.8E+01	1.3E-03	7.7E+01	2.1E-03	1.3E+02	3.6E-03	2.1E+02	5.7E-03	2.2E+02	5.9E-03
Ba-140	2.7E+02	7.3E-03	4.3E+02	1.2E-02	7.5E+02	2.0E-02	1.2E+03	3.2E-02	1.2E+03	3.3E-02
La-140	2.7E+02	7.3E-03	4.3E+02	1.2E-02	7.5E+02	2.0E-02	1.2E+03	3.2E-02	1.2E+03	3.3E-02
Ce-141	1.9E+01	5.3E-04	3.1E+01	8.4E-04	5.4E+01	1.5E-03	8.6E+01	2.3E-03	8.8E+01	2.4E-03
Ce-144	1.9E+00	5.3E-05	3.1E+00	8.4E-05	5.4E+00	1.5E-04	8.6E+00	2.3E-04	8.8E+00	2.4E-04
Np-239	5.4E+03	1.5E-01	8.6E+03	2.3E-01	1.5E+04	4.1E-01	2.4E+04	6.4E-01	2.5E+04	6.6E-01

Table 15.4-19
Instrument Line Break Accident Results

Exposure Location and Time Period/Duration	Maximum Calculated TEDE Sv (REM)	Acceptance Criterion TEDE Sv (REM)
Exclusion Area Boundary (EAB) for any (worst) 2 hour period		
Pre-incident Spike	2.2E-04 (0.022)	0.025 (2.5)
Equilibrium Iodine Activity	3.8E-03 (0.38)	0.25 (25)
Outer Boundary of Low Population Zone (LPZ) for the Duration of the Accident (30 days)		
Pre-incident Spike	3.4E-05 (0.0034)	0.025 (2.5)
Equilibrium Iodine Activity	5.9E-04 (0.059)	0.25 (25)
Control Room Operator Dose for the Duration of the Accident (30 days)		
Pre-incident Spike	1.7E-04 (0.017)	0.05 (5.0)
Equilibrium Iodine Activity	2.8E-03 (0.28)	0.05 (5.0)

15.4 ANALYSIS OF ACCIDENTS

15.4.9 RWCU/SDC System Line Failure Outside Containment

15.4.9.1 Identification of Causes

To evaluate liquid process line pipe breaks outside containment, the failure of a cleanup water line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the cleanup water line, representing the most significant liquid reactor coolant line outside containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

15.4.9.2 Sequence of Events and Systems Operation

~~15.4.9.2.1 Sequence of Events~~

The sequence of events is presented in Table 15.4-20.

~~15.4.9.2.2~~ 15.4.9.2.1 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required or credited in the radiological consequence analysis. However, the operator should perform the following (shown for informational purposes only) actions:

- Determine that a line break has occurred
- Ensure that if vessel water level is below level 3 that reactor has scrammed,
- Confirm RWCU/SDC System containment isolation valves closed,
- Monitor vessel water level and ensure actuation of ECCS as needed, and
- Implement site radiation incident procedures.

These actions occur over an elapsed time of 3–4 hours.

~~15.4.9.2.3~~ 15.4.9.2.2 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the ECCS. The Reactor Protection System, SRVs, ECCS, and safety-related functions of the Control Rod Drive system are assumed to function properly to ensure a safe shutdown.

The ESF systems, including the ADS and GDCS, are assumed to operate normally.

15.4.9.3 Core and System Performance

The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.4.9.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in piping connected to the RCPB or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.4.5. The cleanup water system piping break is less severe than the main steamline break.

15.4.9.5 Radiological Consequences

15.4.9.5.1 ~~General~~Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC.

Specific values of parameters used in the evaluation are presented in Table 15.4-21.

~~15.4.9.5.2 Fission Product Release~~

Source Term: There is no fuel damage as a consequence of this accident. The only activity available for release from the break is that which is present in the reactor coolant and RWCU/SDC System downstream components prior to the break.

Isolation of the line is conservatively analyzed based upon actuation of the flow differential pressure instrumentation. A total of 46 seconds is allowed for differential flow detection and time delay prior to initiating containment isolation valve closure. After the initial 46 seconds, containment isolation valves close over a period of 20 seconds. The initial break flow rate is limited to 2218 kg/sec assuming two-phase critical flow for limiting diameter piping inside containment. The initial break flow rate is assumed to remain constant for the initial 50-seconds following the pipe break. The flow rate is assumed to linearly decrease to zero over the subsequent 16-second period. The total break release period for sources inside containment is 66 seconds.

In addition to the flow of reactor coolant out of the break, the total non-filtered inventory contained in the RWCU/SDC System regenerative and non-regenerative heat exchanger is released. Check valves prevent back flow of inventory from the upstream demineralizer. A break on the downstream side of the demineralizer is bounded by the assumed break location due to reduced flow, steam flashing, and radionuclide source concentrations downstream of the heat exchangers and demineralizer.

Reactor coolant radionuclide source terms are calculated consistent with Regulatory Guide 1.183 iodine spiking assumptions provided for analyzing consequences of a Main Steam Line Break for a BWR as presented in Subsection 15.4.5.5.1. Noble gas activity in the reactor coolant is negligible and is therefore ignored in this analysis.

A summary of RWCU/SDC System line break accident radiological consequence assumptions are provided in Table 15.4-21.

~~**15.4.9.5.3 Fission Product Transport to the Environment**~~

Fission Product Transport to the Environment: It is conservatively assumed that the release to the environment via the Reactor Building is instantaneous, with no iodine plateout. ~~The release location is the Reactor Building.~~ No credit is taken for holdup in the Reactor Building. Separate flashing fractions are applied to each reactor coolant release source as presented in Table 15.4-21. Fission product releases to the environment are presented in Table 15.4-22.

Control Room: Control Room ventilation is assumed to operate in the normal mode for the duration of the event. No credit is taken for Control Room isolation, or operation of the Control Room emergency filter units (EFU).

~~**15.4.9.5.4 Assumptions Requiring Confirmation**~~

Assumptions Requiring Confirmation: Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

~~15.4.9.5.5~~ 15.4.9.5.2 Results

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

Table 15.4-21

RWCU/SDC Line Break Accident Parameters

I. Data and assumptions used to estimate source terms	
A. Fuel Damage	none
B. Reactor Coolant Activity, Bq/g (µCi/g) DE I-131~~:~~ Pre-incident Spike Equilibrium Iodine Activity	 148,000 (4.0) ~~µCi/g DE I-131~~ 7400 (0.2) ~~µCi/g DE I-131~~
C. Water Mass Released, kg (lbm) RPV Coolant Blow-down RWCU/SDC System RHX RWCU/SDC System NRHX	 128,650 (283,620) 975 (2150) 3651 (8050)
II. Data and assumptions used to estimate activity released	
A. Water-to-Steam Flashing Fractions RPV Coolant Blow-down RWCU/SDC System RHX RWCU/SDC System NRHX	 0.38 0.28 0.074
B. Iodine Plateout Fraction, %	0
C. Reactor Building Flow rate, %/hour	Instantaneous
III. Control Room Parameters	
A. Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E4)
B. Unfiltered intake, l/s (ft^3/min)	200 (424)
C. Filtered intake, l/s (ft^3/min)	0 (0)
D. Unfiltered inleakage, l/s (ft^3/min)	0 (0)
E. Occupancy Factors	
0 – 1 day	1.0
1 – 4 days	0.6
4 – 30 days	0.4

Table 15.4-21

RWCU/SDC Line Break Accident Parameters

~~III~~ IV. Dispersion and Dose Data	
A. Offsite Meteorology ~~EAB~~ ~~LPZ~~ ~~Control Room~~ ~~Reactor Building Release~~	 ~~2.00E-03 s/m³~~ ~~Table 2.0-1~~ ~~Table 2.0-1~~
Exclusion Area Boundary	
0 – 2 hrs	2.00E-03 s/m³
Low Population	
0 – 8 hrs	1.90E-04 s/m³
> 8 hrs	NR*
B. Control Room Meteorology (Reactor Building Release Point)	
0 – 2 hrs	1.50E-03 s/m³
> 2 hrs	NR*
~~B~~C. Method of Dose Calculation	RG 1.183
~~C~~D Dose conversion Assumptions	RG 1.183
~~D~~E. Activity Inventory/Releases	Table 15.4-22
~~E~~F. Dose Evaluations	Table 15.4-23

* Due to the short release, values > 2 hours do not impact the calculated doses and are Not Required (NR).

Table 15.4-22

RWCU/SDS Line Break Accident Isotopic Release to Environment

Isotope	~~Coincident Spike~~Equilibrium Iodine ~~(MBq)~~		Pre-incident Spike ~~(MBq)~~	
	MBq	Ci	MBq	Ci
I-131	1.52E+05~~1.46E+05~~	4.12E+00	3.05E+06~~2.92E+06~~	8.23E+01
I-132	1.37E+06~~1.38E+06~~	3.71E+01	2.74E+07~~2.77E+07~~	7.41E+02
I-133	9.97E+05~~1.01E+06~~	2.69E+01	1.99E+07~~2.02E+07~~	5.39E+02
I-134	2.49E+06~~2.54E+06~~	6.74E+01	4.99E+07~~5.09E+07~~	1.35E+03
I-135	1.38E+06~~1.42E+06~~	3.74E+01	2.77E+07~~2.84E+07~~	7.49E+02
Cs-134	1.01E+03~~1.68E+03~~	2.73E-02	1.01E+03~~3.37E+04~~	2.73E-02
Cs-136	6.72E+02~~1.12E+03~~	1.82E-02	6.72E+02~~2.24E+04~~	1.82E-02
Cs-137	2.69E+03~~4.49E+03~~	7.27E-02	2.69E+03~~8.97E+04~~	7.27E-02
Co-58	8.24E+02~~8.40E+02~~	2.23E-02	8.24E+02~~8.40E+02~~	2.23E-02
Co-60	1.63E+03~~1.63E+03~~	4.40E-02	1.63E+03~~1.63E+03~~	4.40E-02
Sr-89	3.78E+03~~6.36E+03~~	1.02E-01	3.78E+03~~1.27E+05~~	1.02E-01
Sr-90	2.60E+02~~4.49E+02~~	7.04E-03	2.60E+02~~8.97E+03~~	7.04E-03
Y-90	2.60E+02~~4.49E+02~~	7.04E-03	2.60E+02~~8.97E+03~~	7.04E-03
Sr-91	1.43E+05~~2.39E+05~~	3.86E+00	1.43E+05~~4.79E+06~~	3.86E+00
Sr-92	3.44E+05~~5.61E+05~~	9.31E+00	3.44E+05~~1.12E+07~~	9.31E+00
Y-91	1.51E+03~~2.47E+03~~	4.09E-02	1.51E+03~~4.94E+04~~	4.09E-02
Y-92	2.10E+05~~3.48E+05~~	5.68E+00	2.10E+05~~6.95E+06~~	5.68E+00
Y-93	1.43E+05~~2.39E+05~~	3.86E+00	1.43E+05~~4.79E+06~~	3.86E+00

Table 15.4-22

RWCU/SDS Line Break Accident Isotopic Release to Environment

Isotope	~~Coincident Spike~~Equilibrium Iodine ~~(MBq)~~		Pre-incident Spike ~~(MBq)~~	
	MBq	Ci	MBq	Ci
Zr-95	3.02E+02~~4.86E+02~~	8.18E-03	3.02E+02~~9.72E+03~~	8.18E-03
Nb-95	3.02E+02~~4.86E+02~~	8.18E-03	3.02E+02~~9.72E+03~~	8.18E-03
Mo-99	7.48E+04~~1.23E+05~~	2.02E+00	7.48E+04~~2.47E+06~~	2.02E+00
Tc-99m	7.48E+04~~1.23E+05~~	2.02E+00	7.48E+04~~2.47E+06~~	2.02E+00
Ru-103	7.48E+02~~1.23E+03~~	2.02E-02	7.48E+02~~2.47E+04~~	2.02E-02
Ru-106	1.09E+02~~1.87E+02~~	2.95E-03	1.09E+02~~3.74E+03~~	2.95E-03
Te-129m	1.51E+03~~2.47E+03~~	4.09E-02	1.51E+03~~4.94E+04~~	4.09E-02
Te-131m	3.70E+03~~5.98E+03~~	9.99E-02	3.70E+03~~1.20E+05~~	9.99E-02
Te-132	3.78E+02~~5.98E+02~~	1.02E-02	3.78E+02~~1.20E+04~~	1.02E-02
Ba-140	1.51E+04~~2.47E+0~~4	4.09E-01	1.51E+04~~4.94E+05~~	4.09E-01
La-140	1.51E+04~~2.47E+0~~4	4.09E-01	1.51E+04~~4.94E+05~~	4.09E-01
Ce141	1.09E+03~~1.87E+03~~	2.95E-02	1.09E+03~~3.74E+04~~	2.95E-02
Ce-144	1.09E+02~~1.87E+02~~	2.95E-03	1.09E+02~~3.74E+03~~	2.95E-03
Np-239	3.02E+05~~4.86E+05~~	8.18E+00	3.02E+05~~9.72E+06~~	8.18E+00

Table 15.4-23

RWCU/SDC Line Break Accident Results

Exposure Location and Time Period/Duration	Maximum Calculated TEDE, Sv (~~rem~~REM)	Acceptance Criterion TEDE, ~~(rem~~Sv~~)~~
Exclusion Area Boundary (EAB) for any (Worst) 2-hour Period~~the Entire Period of the Radioactive Cloud Passage~~		
Equilibrium~~Coincident~~ Iodine Spike Case	0.0045 (0.~~49~~45)	~~2.5~~0.025
Pre-incident Iodine Spike Case	0.076 (~~9.8~~7.6)	~~25~~0.25
Outer Boundary of Low Population Zone (LPZ) for Duration of the Accident (30 days)~~the Entire Period of the Radioactive Cloud Passage~~		
Equilibrium~~Coincident~~ Iodine Spike Case	4.3E-04 (0.~~047~~043)	~~2.5~~0.025
Pre-incident Iodine Spike Case	0.0073 (0.~~93~~73)	~~25~~0.25
Control Room Dose for the Duration of the Accident (30 days)		
~~Coincident~~ Equilibrium Iodine Spike Case	0.0022 (0.~~24~~22)	~~5~~0.05
Pre-incident Iodine Spike Case	0.035 (~~4.7~~3.5)	~~5~~0.05