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MFN 07-100, Supplement 1
and 07-456

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Subject: **Response to Portion of NRC Request for Additional Information Letter Nos. 69 and 100 Related to ESBWR Design Certification Application – Safety Analyses – RAI Numbers 6.2-168 and 15.4-2S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letters dated October 11, 2006 and May 30, 2007. GEH responses to RAI Numbers 6.2-168 and 15.4-2S01 are addressed in Enclosure 1. The DCD Markups are addressed in Enclosure 2.

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

Sincerely,

Kathy Sedney for

James C. Kinsey
Vice President, ESBWR Licensing

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NRO

References:

1. MFN 06-381 – Letter from US Nuclear Regulatory Commission (NRC) to David H. Hinds, *Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application,* dated October 11, 2006
2. MFN 07-327 – Letter from US Nuclear Regulatory Commission (NRC) to Robert E. Brown, *Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application*, dated May 30, 2007

Enclosures:

1. Response to NRC Request for Additional Information Letter Nos. 69 and 100 Related to ESBWR Design Certification Application – Safety Analyses, RAI Numbers 6.2-168 and 15.4-2S01
2. MFNs 07-100, Supplement 1 and 07-456 DCD Markups

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRF	0064-2601, Revision 1

Enclosure 1

MFNs 07-100, Supplement 1 and 07-456

Response to Portion of NRC Request for Additional Information Letter Nos. 69 and 100 Related to ESBWR Design Certification Application

Safety Analyses

RAI Numbers 6.2-168 and 15.4-2 S01

NRC RAI 6.2-168:

In DCD, Tier 2, Revision 3, Section 6.2.3, the applicant states because the containment is located entirely within the reactor building (RB), multiple structural barriers exist between the containment and the environment. Therefore, fission product leakage from the RB is mitigated. The staff is reviewing the degree of mitigation provided by the RB in connection with RAI's on mixing assumptions and building leakage. Building leakage is especially important because of its impact on the effectiveness of the building being a barrier to the release of radioactivity to the environment. DCD, Tier 1, Section 2.16.5 states that "offsite dose requirements are met assuming a 100 percent volume change out per day in the RB volume outside of the RCCV." This is inconsistent with the assumption used in the design basis analyses of 50 percent volume per day in Chapter 15, Table 15.4-5. Please explain the difference and make corrections to the appropriate sections of the DCD.

GEH Response:

Reactor Building.(RB) leakage is assumed to be 50 wt.% per day in the Loss of Coolant Accident dose consequence analysis, as documented in DCD Tier 2, Subsection 15.4.4. DCD, Tier 2, Subsection 6.2.3 has been revised to reflect the 50 wt.% per day through the RB as indicated on the attached markups. Subsections 6.2.1 and 6.2.4 were also revised for compliance to 10 CFR 50.34(a)(1) rather than 10 CFR 50.67.

DCD Impact:

DCD Tier 2, Subsections 6.2.1, 6.2.3, and 6.2.4 will be revised as noted on the attached markup in Revision 5. Please note Tier 1 Section 2.16.5, Table 2.16.5-2, Item 4, DCD Revision 4 currently has the reactor building leakage rate of 50% wt. per day.

NRC RAI 15.4-2 S01:

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

(1) Provide revised steam and water mass releases for the main steam line break accident.

(2) Add the following information to Table 15.4-11: 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

(3) Revise the following information in Table 15.4-13:

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.

c) Provide control room operator doses for pre and post-iodine spike.

GEH Response:

(1) The mass releases for a MSLB outside of containment (hot standby) for the ESBWR are as follows:

- **Liquid:** 34547 kg (82328 kg assumed in DCD, Revision 3 MSLB analysis)
- **Steam:** 16295 kg (4705 kg assumed in DCD, Revision 3 MSLB analysis)

Since the steam release assumed in the DCD, Revision 3 analysis did not bound the calculated ESBWR steam release, the dose consequences will be revised and the results will be reflected in DCD, Tier 2, Revision 5. The revised analysis includes an additional 25% liquid and steam mass to account for the final optimized design of the main steam lines, main steam isolation valves and venturis.

- **Liquid:** 43184 kg (92204 lbm)
- **Steam:** 20369 kg (44905 lbm)

See attached DCD, Subsection 15.4.5 markups.

(2) The duration of the accident for EAB, LPZ and control room X/Q values, release point, control room operator doses, control room is not isolated during the event, and the control room ventilation system is in normal operation will be added to Table 15.4-11. See attached DCD markups.

(3) Table 15.4-13 will be revised to clarify that the EAB cloud passage is considered for the worst 2 hours, the LPZ dose is considered for 0 to 30 days, and add the control room doses to the operators pre and post-iodine spike. See attached DCD markups.

Enclosure 2

MFNs 07-100, Supplement 1 and 07-456

DCD Markups

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Pressure Suppression Containment

Relevant to ESBWR pressure suppression containment system, this subsection addresses or references to other DCD locations that address the applicable requirements of GDC 4, 16, 50, and 53 discussed in Standard Review Plan (SRP) 6.2.1.1.C Rev. 6. The plant meets the requirements of

(1) GDC 4, as it relates to the environmental and missile protection design, requires that safety-related structures, systems, and components be designed to accommodate the dynamic effects (for example, effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during normal plant operation or following a loss-of-coolant accident;

(2) GDC 16 and 50, as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident; and

(3) GDC 53 as it relates to the containment design capabilities provided to ensure that the containment design permits periodic inspection, an appropriate surveillance program, and periodic testing at containment design pressure.

6.2.1.1.1 Design Bases

The pressure suppression containment system, which comprises the Drywell (DW) and Wetwell (WW) and supporting systems, is designed to meet the following Safety Design Bases:

- The containment structure shall maintain its functional integrity during and following the peak transient pressures and temperatures, which would occur following any postulated LOCA. A DBA is defined as the worst pipe break, which leads to maximum DW and WW pressure and/or temperature, and is postulated to occur simultaneously with loss of preferred power. For structural integrity evaluation, Safe Shutdown Earthquake (SSE) loads are combined with LOCA loads.
- The containment structure design shall accommodate the full range of loading conditions consistent with normal plant operation, Safety Relief Valve (SRV) discharge and accident conditions including the LOCA related design loads.
- The containment structure is designed to accommodate the maximum internal negative pressure difference between DW and WW, and the maximum external negative pressure difference relative to the RB surrounding the containment.
- The containment structure and RB, with concurrent operation of containment isolation function (isolates all pipes or ducts which penetrate the containment boundary) and other accident mitigation systems, shall limit fission product leakage during and following the postulated DBA to values less than leakage rates which would result in off-site doses greater than those set forth in 10 CFR 50.~~67~~34(a)(1).

6.2.2.4 Testing and Inspection Requirements

The PCCS is an extension of the containment, and it will be periodically pressure tested as part of overall containment pressure testing (Section 6.2.6). Also, the PCCS loops can be isolated for individual pressure testing during maintenance.

If additional inservice inspection becomes necessary, it is unnecessary to remove the PCCS condenser because ultrasonic testing of tube-to-header welds and eddy current testing of tubes can be done with the PCCS condensers in place during refueling outages.

6.2.2.5 Instrumentation Requirements

The PCCS does not have instrumentation that is separate from the Containment System. Control logic is not needed for it's functioning. There are no sensing and power actuated devices. Containment System instrumentation is described in Subsection 6.2.1.7.

6.2.3 Reactor Building Functional Design

Relevant to the function of a secondary containment design, this subsection addresses (or references to other DCD locations that address) the applicable requirements of GDC 4, 16, and 43 and Appendix J to 10 CFR 50 discussed in SRP 6.2.3 R2. The plant meets the relevant and applicable requirements of:

- GDC 4 as it relates to safety-related structures, systems and components being designed to accommodate the effects of normal operation, maintenance, testing and postulated accidents, and being protected against dynamic effects (for example, the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures;
- GDC 16 as it relates to reactor containment and associated systems being provided to establish an essentially leak-tight barriers against the uncontrolled release of radioactive material to the environment;
- GDC 43 as it relates to atmosphere cleanup systems having the design capability to permit periodic functional testing to ensure system integrity, the operability of active components, and the operability of the system as a whole and the performance of the operational sequence that brings the system into operation; and
- 10 CFR 50, Appendix J as it relates to the secondary containment being designed to permit preoperational and periodic leakage rate testing so that bypass leakage paths are identified.

This subsection applies to the ESBWR RB design. The RB structure encloses penetrations through the containment (except for those of the main steam tunnel and IC/PCC pools). The RB:

- Provides an added barrier to fission product released from the containment in case of an accident;
- Contains, dilutes, and holds up any leakage from the containment; and
- Houses safety-related systems.

The RB under accident conditions is automatically isolated to provide a hold up ~~and plate out barrier~~volume for fission products. When isolated, the RB can be serviced by the RB HVAC

system through a High Efficiency Particulate Air/Absolute (HEPA) filtration system (Refer to Subsection 9.4.6) No credit is taken for the filters in dose consequence analyses (See Section 15.4.4). With low leakage and stagnant conditions, ~~hold up and plate out mechanisms perform~~ the basic mitigating functions is the hold up of fission products in the RB itself. The ESBWR design does not include a secondary containment; ~~and minimal~~however credit is taken for the existence of the RB surrounding the primary containment vessel in ~~any~~ radiological analyses. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.54% per day and a RB ~~bypass~~ leakage~~,~~ ~~equal to~~rate of 50% per day ~~100% of the containment leak rate~~, show that off-site and control room doses after an accident are less than allowable limits, as discussed in Chapter 15. The RB envelope is not intended to provide a leak-tight barrier against radiological releases. Therefore, the design criterion of GDC 16 does not apply.

During normal plant operation, potentially contaminated areas within the RB are kept at a negative pressure with respect to the environment while clean areas are maintained at positive pressure. The ESBWR does not need, and thus has no filter system that performs a safety-related function following a design basis accident, as discussed in Subsection 6.5.1. Therefore the design criterion of GDC 43 is not applicable.

Personnel and equipment entrances to the RB consist of vestibules with interlocked doors and hatches. Large equipment access is by means of a dedicated, external access tower that provides the necessary interlocks.

6.2.3.4 Design Bases

The RB is designed to meet the following safety design bases:

- The RB maintains its integrity during the environmental conditions postulated for a DBA;
- The RB HVAC system automatically isolates upon detection of high radiation levels in the ventilation exhaust system;
- Openings through the RB boundary, such as personnel and equipment doors, are closed during normal operation and after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms that are monitored in the control room.
- Detection and isolation capability for high-energy pipe breaks within the RB is provided;
- The compartments within the RB are designed to withstand the maximum pressure due to a High-Energy Line Break (HELB). Each line break analyzed is a double-ended break. In this analysis, the rupture producing the greatest blowdown of mass and enthalpy in conjunction with worst-case single active component failure is considered. Blowout panels between compartments provide flow paths to relieve pressure.
- The RB is capable of periodic testing to assure that the leakage rates assumed in the radiological analyses are met.

6.2.3.5 Design Description

The RB is a reinforced concrete structure that forms an envelope completely surrounding the containment (except the basemat). The boundary of the clean areas and the RB are shown in Figure 6.2-17.

The criteria for the design of the LD&IS, which provides containment and reactor vessel isolation control, are listed in Subsection 7.1.2. The bases for assigning certain signals for containment isolation are listed and explained in Subsection 7.3.3.

6.2.4.2 System Design

The containment isolation function is accomplished by valves and control signals, required for the isolation of lines penetrating the containment. The RCPB influent lines are identified in Table 6.2-13, and the RCPB effluent lines are identified in Table 6.2-14. Table 6.2-15 through 6.2-42 show the pertinent data for the containment isolation valves. A detailed discussion of the LD&IS controls associated with the containment isolation function is included in Subsection 7.3.3.

Power-operated containment isolation valves have position indicating switches in the control room to show whether the valve is open or closed. Power for valves used in series originates from physically independent sources without cross ties to assure that no single event can interrupt motive power to both closure devices.

All POVs with geared or bi-directional actuators (motorized or fluid-powered) remain in their last position upon failure of valve power. All POVs with fluid-operated/spring-return actuators (not applicable to air-testable check valves) close on loss of fluid pressure or power supply. To support the inerted containment design, pneumatic actuators for valves located inside containment are supplied with pressurized nitrogen gas, whereas pneumatic actuators for valves located outside of containment are generally supplied compressed air.

The design of the containment isolation function includes consideration for possible adverse effects of sudden isolation valve closure when the plant systems are functioning under normal operation.

General compliance or alternate approach assessment for Regulatory Guide 1.26 may be found in Subsection 3.2.2. General compliance or alternate approach assessment for Regulatory Guide 1.29 may be found in Subsection 3.2.1.

Containment isolation valves are generally automatically actuated by the various signals in primary actuation mode or are remote-manually operated in secondary actuation mode. Other appropriate actuation modes, such as process-actuated check valves, are identified in the containment isolation valve information Tables 6.2-13 through 6.2-42.

6.2.4.2.1 Containment Isolation Valve Closure Times

Containment isolation valve closure times are established by determining the isolation requirements necessary to keep radiological effects from exceeding guidelines in ~~10 CFR50.67~~10 CFR 50.34(a)(1). For system lines, which can provide an open path from the containment to the environment, a discussion of valve closure time bases is provided in Chapter 15. However the design values of closure times for power-operated valves is more conservative than the above requirement. For valves above 80 mm (3 inches) up to and including 300 mm (12 inches) in diameter, the closure time is at least within a time determined by dividing the nominal valve diameter by 300 mm (12 inches) per minute. Valves 80 mm (3 inches) and less generally close within 15 seconds. All valves larger than 300 mm (12 inches) in diameter close within 60

15.4.5.4 Barrier Performance

Because this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3.

Initially, only steam issues from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter for each line. Rapid depressurization of the RPV causes the water level to rise, resulting in a steam-water mixture flowing from the break until the valves are closed. The total integrated mass leaving the RPV through the steamline break is provided in Table 15.4-11.

15.4.5.5 Radiological Consequences

The radiological analysis for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet Regulatory Guide 1.183 and 10 CFR 50.34(a)(1) guidelines. This analysis is referred to as the "design basis analysis."

15.4.5.5.1 Design Basis Analysis

Specific values of parameters used in the evaluation are presented in Table 15.4-11.

General Compliance or Alternate Approach Statement (RG 1.183): This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a MSLBA for a BWR.

Some of the models and conditions that are prescribed are inconsistent with actual physical phenomena. The effect of the conservative bias that is introduced is generally limited to plant design choices not within the scope of the ESBWR Standard Plant design. The resultant dose is within regulatory limits.

Source Term: There is no fuel damage as a result of this accident. The only activity available for release from the break is that which is present in the reactor coolant and steamlines prior to the break.

Since there is no fuel damage, the source term is based on the design basis concentrations for steam and water. Iodine isotopes (Table 11.1-4a) are adjusted to account for the maximum equilibrium iodine and pre-incident iodine spike concentrations (7400 Bq/g [0.2 μCi/g] and 148000 Bq/g [4.0 μCi/g], respectively) in accordance with Regulatory Guide 1.183 guidance. The design basis source term is based on an assumed offgas release rate of 100,000 μCi/second after 30 minutes of decay. Branch Technical Position 11-5 of the Standard Review Plan lists a value of 100 μCi/second per MWt after 30 minutes decay. The noble gas source term presented in Table 11.1-2a is conservatively adjusted accordingly,

$$\frac{102\% \times 4500 MWt \times 100\,{}^{\mu Ci}/_{s-MWt}}{100000\,{}^{\mu Ci}/_{s}} = 4.59\,.$$

The increase in iodine concentration could occur from additional minor leakage from the fuel. These increases would not have a significant impact on activation products (Co-58, co-60, etc.), therefore no adjustment is warranted for those isotopes. The remaining isotopes (Table 11.1-5a)

are conservatively adjusted by a factor of 4.59 as well. The activity released to the environment as a result of a MSLBA is presented in Table 15.4-12.

Fission Product Transport to the Environment: The transport pathway is a direct unfiltered release to the environment with a release rate of 1.0E+08 weight % per day. The release location is the Turbine Building. No credit is taken for holdup in the Turbine Building. All of the activity in the steam is released to the environment. A flashing fraction of 0.4 is applied to the reactor coolant released. .

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).

Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.5.5.2 Results

The calculated exposures for the design basis analysis are presented in Table 15.4-13 and are less than the guidelines of RG 1.183 and 10 CFR 50.34(a)(1).

15.4.6 Control Rod Drop Accident

15.4.6.1 Features of the ESBWR Fine Motion Control Rod Drives

As presented in Subsection 4.6.1, the Fine Motion Control Rod Drive (FMCRD) has several new features that are unique compared with locking piston control rod drives.

In each FMCRD, there are dual safety-related separation-detection devices that detect the separation of the control rod from the FMCRD if the control rod is stuck and separated from the ballnut of the FMCRD. The control rods are normally inserted into the core and withdrawn with the hollow piston, which is connected with the control rod, resting on the ballnut. The separation-detection device is used at all times to ascertain that the hollow piston and control rod are resting on the ballnut of the FMCRD. The separation-detection devices sense motion of a spring loaded support for the ball screw and in turn the hollow piston and the control rod. Separation of either the control rod from the hollow piston or the hollow piston from the ballnut is detected immediately. When separation has been detected, the interlocks preventing rod withdrawal operate to prevent further control rod withdrawal. Also, an alarm signal would be initiated in the control room to warn the operator.

There is also the unique highly reliable bayonet type coupling between the control rod blade and the FMCRD. With this coupling, the connection between the blade and the drive cannot be separated unless they are rotated 45 degrees. This rotation is not possible during reactor operation. There are procedural coupling checks to assure proper coupling. Finally, there is the latch mechanism on the hollow piston part of the drive. If the hollow piston is separated from the ballnut and rest of the drive due to stuck rod, the latch limits any subsequent rod drop to a short distance. More detailed descriptions of the FMCRD system are presented in Subsection 4.6.1. Failure modes of the FMCRD are discussed in Appendix 15A.

Table 15.4-11

MSLBA 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) 7,400 (0.2)
C.	Steam Mass Released, kg (lbm)	20,369 (44905)
D.	Water Mass Released, kg (lbm)	43,184 (95204)
E.	ESBWR Off-gas Design Basis Release Rate, MBq/s (μCi/s)	3,700 (100,000)
F.	SRP BTP 11-5 Off-gas Release Rate, MBq/s (μCi/s)	3.7 (100)
G.	Water Flashing Fraction	0.4
II.	Data and assumptions used to estimate activity released	
A.	Isolation valve closure time, sec	5
B.	MSIV Response time, sec	0.5
C.	Total assumed release duration, sec	5.5
III	Control Room Parameters	
A.	Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E4)
B.	Unfiltered intake, l/s (cfm)	200 (424)
C.	Filtered intake, l/s (cfm)	0 (0)
D.	Unfiltered inleakage, l/s (cfm)	0 (0)
E.	Occupancy Factors	
	0 – 1 day	1.0
	1 – 4 days	0.6
	4 – 30 days	0.4
IV.	Dispersion Data	
A.	Off-site Meteorology	
	Exclusion Area Boundary	
	0 – 2 hrs	2.00E-03 s/m^3

Low Population	
0 – 8 hrs	1.90E-04 s/m^3
> 8 hrs	NR*
Exclusion Area Boundary	
B. Control Room Meteorology (Turbine Building Release Point)	
0 – 2 hrs	1.20E-03 s/m^3
> 2 hrs	NR*
C. Method of Dose Calculation	RG 1.183
D. Dose Conversion Assumptions	RG 1.183
E. Activity Inventory and Releases	Tables 15.4-12
F. Dose Evaluations	Table 15.4-13

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

Table 15.4-12
MSLBA Environment Releases

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
Co-58	2.9E+02	7.8E-03	2.9E+02	7.8E-03
Co-60	5.7E+02	1.5E-02	5.7E+02	1.5E-02
Kr-85	3.4E+01	9.2E-04	3.4E+01	9.2E-04
Kr-85m	8.6E+03	2.3E-01	8.6E+03	2.3E-01
Kr-87	2.8E+04	7.6E-01	2.8E+04	7.6E-01
Kr-88	2.8E+04	7.6E-01	2.8E+04	7.6E-01
Rb-86	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sr-89	1.3E+03	3.6E-02	1.3E+03	3.6E-02
Sr-90	9.1E+01	2.5E-03	9.1E+01	2.5E-03
Sr-91	5.0E+04	1.3E+00	5.0E+04	1.3E+00
Sr-92	1.2E+05	3.3E+00	1.2E+05	3.3E+00
Y-90	9.1E+01	2.5E-03	9.1E+01	2.5E-03
Y-91	5.3E+02	1.4E-02	5.3E+02	1.4E-02
Y-92	7.3E+04	2.0E+00	7.3E+04	2.0E+00
Y-93	5.0E+04	1.3E+00	5.0E+04	1.3E+00
Zr-95	1.1E+02	2.9E-03	1.1E+02	2.9E-03
Zr-97	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Nb-95	1.1E+02	2.9E-03	1.1E+02	2.9E-03
Mo-99	2.6E+04	7.1E-01	2.6E+04	7.1E-01
Tc-99m	2.6E+04	7.1E-01	2.6E+04	7.1E-01
Ru-103	2.6E+02	7.1E-03	2.6E+02	7.1E-03
Ru-105	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ru-106	3.8E+01	1.0E-03	3.8E+01	1.0E-03
Rh-105	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sb-127	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sb-129	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-127	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-127m	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-129	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-129m	5.3E+02	1.4E-02	5.3E+02	1.4E-02
Te-131m	1.3E+03	3.5E-02	1.3E+03	3.5E-02
Te-132	1.3E+02	3.6E-03	1.3E+02	3.6E-03
I-131	5.4E+04	1.5E+00	1.1E+06	2.9E+01
I-132	4.9E+05	1.3E+01	9.8E+06	2.6E+02
I-133	3.6E+05	9.6E+00	7.1E+06	1.9E+02
I-134	8.9E+05	2.4E+01	1.8E+07	4.8E+02
I-135	5.0E+05	1.3E+01	9.9E+06	2.7E+02
Xe-133	1.2E+04	3.2E-01	1.2E+04	3.2E-01
Xe-135	3.3E+04	8.8E-01	3.3E+04	8.8E-01

Table 15.4-12
MSLBA Environment Releases

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
Cs-134	3.5E+02	9.5E-03	3.5E+02	9.5E-03
Cs-136	2.3E+02	6.4E-03	2.3E+02	6.4E-03
Cs-137	9.4E+02	2.5E-02	9.4E+02	2.5E-02
Ba-139	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ba-140	5.3E+03	1.4E-01	5.3E+03	1.4E-01
La-140	5.3E+03	1.4E-01	5.3E+03	1.4E-01
La-141	0.0E+00	0.0E+00	0.0E+00	0.0E+00
La-142	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-141	3.8E+02	1.0E-02	3.8E+02	1.0E-02
Ce-143	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-144	3.8E+01	1.0E-03	3.8E+01	1.0E-03
Pr-143	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Nd-147	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Np-239	1.1E+05	2.9E+00	1.1E+05	2.9E+00
Pu-238	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pu-239	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pu-240	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pu-241	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Am-241	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cm-242	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cm-244	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table 15.4-13

MSLBA Analysis Results

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 Activity	0.0016 (0.16)	0.025
Pre-incident Spike	0.0271 (2.71)	0.25
Outer Boundary of Low Population Zone (LPZ) for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	<0.001 (<0.1)	0.025
Pre-incident Spike	0.0026 (0.26)	0.25
Control Room Operator Dose for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	<0.001 (<0.1)	0.05
Pre-incident Spike	0.01 (1.0)	0.05

15.4.4.5.4 Meteorology and Site Assumptions

Offsite Meteorology - This DCD uses a generic U.S. site that does not specifically identify meteorological parameters adequate to define dispersion conditions for accident evaluation. Therefore, a set of dispersion parameters (χ/Q's) were selected to simulate a U.S. site, which are given in Table 15.4-9 for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ).

Control Room Meteorology - No specific acceptable method exists to calculate the meteorology for standard plant application for control room dose analysis. The control room assumed dispersion factors (χ/Q) are provided in Table 15.4-9.

15.4.4.5.5 Breathing Rates

The breathing rates assumed in the analysis presented in Table 15.4-5. These values are consistent with RG 1.183, Section 4.1.3.

15.4.4.6 Results

The results of this analysis are presented in Table 15.4-9 for both offsite and control room dose evaluations and are within 10 CFR 50.34 and RG 1.183 regulatory guidelines. The following criteria are met:

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(3) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

15.4.4.6.1 Assumptions to be Confirmed by the COL Applicant

- The assumption in the radiological analysis that require confirmation by the COL Applicant are documented in Section 15.4.11.

15.4.4.6.2 (Deleted)

15.4.5 Main Steamline Break Accident Outside Containment

This event involves postulating a large steam line pipe break outside containment. It is assumed that the largest steam line instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of all main steamlines including the broken line and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside containment.

The Main Steamline Break Accident (MSLBA) containment response evaluation is provided in Section 6.2.

The MSLB ECCS capability evaluation is provided in Section 6.3.

The MSLB radiological evaluation is as follows:

15.4.5.1 Identification of Causes

A MSLBA is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steam line is assumed to occur.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the result of postulated breaches in 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 event for breaks outside the containment is a complete severance of one of the main steamlines. The sequence of events and approximate time required to reach the event is given in Table 15.4-10.

Following isolation of the main steam supply system (i.e., MSIV closure), the ADS initiates automatically on low water level (Level 1). Once the reactor system has depressurized, the GDCS automatically begins reflooding the reactor vessel. The core remains covered throughout the accident, and there is no fuel damage.

15.4.5.2.2 Systems Operation

A postulated guillotine break of one of the main steam lines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle. Flow from the downstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle for the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

A discussion of plant and RPS action and ESF action is presented in Sections 6.3, 7.3 and 7.6.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

The steamline break outside the containment is a special case of the general LOCA break spectrum considered in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in Subsection 6.3.3. For the steamline break outside the containment, the worst single failure does not result in core uncovery (see Section 6.3 for analysis details).

15.4.5.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are presented in Section 6.3. The temperature and pressure transient results from this accident are not sufficient to cause fuel damage.

15.4.5.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the ECCS performance analysis of this event are presented in Table 6.3-1.

15.4.5.3.2 Results

There is no fuel damage as a result of this accident. Refer to Section 6.3 for ECCS analysis.

15.4.5.4 Barrier Performance

Because this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3.

Initially, only steam issues from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter for each line. Rapid depressurization of the RPV causes the water level to rise, resulting in a steam-water mixture flowing from the break until the valves are closed. The total integrated mass leaving the RPV through the steamline break is provided in Table 15.4-11.

15.4.5.5 Radiological Consequences

The radiological analysis for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet Regulatory Guide 1.183 and 10 CFR 50.34(a)(1) guidelines. This analysis is referred to as the "design basis analysis."

15.4.5.5.1 Design Basis Analysis

Specific values of parameters used in the evaluation are presented in Table 15.4-11.

General Compliance or Alternate Approach Statement (RG 1.183): This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a MSLBA for a BWR.

Some of the models and conditions that are prescribed are inconsistent with actual physical phenomena. The effect of the conservative bias that is introduced is generally limited to plant design choices not within the scope of the ESBWR Standard Plant design. The resultant dose is within regulatory limits.

Fission Product Release from Fuel: There is no fuel damage as a result of this accident. The only activity available for release from the break is that which is present in the reactor coolant and steamlines prior to the break.

Fission Product Transport to the Environment: The transport pathway is a direct unfiltered release to the environment with ~~an air exchange rate in~~ a release rate ~~from the Turbine Building~~ of 1.0E+08 ~~volume~~ weight % per day. The release location is the Turbine Building. No credit is

taken for holdup in the Turbine Building. Assuming that all of the activity in the steam becomes airborne, the release of activity to the environment is presented in Table 15.4-12.

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).

Assumptions to be Confirmed by the COL Applicant

The assumptions in the radiological analysis that require confirmation by the COL Applicant are documented in Section 15.4.11.

15.4.5.5.2 Results

The calculated exposures for the design basis analysis are presented in Table 15.4-13 and are less than the guidelines of RG 1.183 and 10 CFR 50.34(a)(1).

15.4.6 Control Rod Drop Accident

15.4.6.1 Features of the ESBWR Fine Motion Control Rod Drives

As presented in Subsection 4.6.1, the Fine Motion Control Rod Drive (FMCRD) has several new features that are unique compared with locking piston control rod drives.

In each FMCRD, there are dual safety-related separation-detection devices that detect the separation of the control rod from the FMCRD if the control rod is stuck and separated from the ballnut of the FMCRD. The control rods are normally inserted into the core and withdrawn with the hollow piston, which is connected with the control rod, resting on the ballnut. The separation-detection device is used at all times to ascertain that the hollow piston and control rod are resting on the ballnut of the FMCRD. The separation-detection devices sense motion of a spring loaded support for the ball screw and in turn the hollow piston and the control rod. Separation of either the control rod from the hollow piston or the hollow piston from the ballnut is detected immediately. When separation has been detected, the interlocks preventing rod withdrawal operate to prevent further control rod withdrawal. Also, an alarm signal would be initiated in the control room to warn the operator.

There is also the unique highly reliable bayonet type coupling between the control rod blade and the FMCRD. With this coupling, the connection between the blade and the drive cannot be separated unless they are rotated 45 degrees. This rotation is not possible during reactor operation. There are procedural coupling checks to assure proper coupling. Finally, there is the latch mechanism on the hollow piston part of the drive. If the hollow piston is separated from the ballnut and rest of the drive due to stuck rod, the latch limits any subsequent rod drop to a short distance. More detailed descriptions of the FMCRD system are presented in Subsection 4.6.1. Failure modes of the FMCRD are discussed in Appendix 15A.

15.4.6.2 Identification of Causes

For the rod drop accident with a potentially adverse result to occur, it is necessary for the following highly unlikely events to occur:

(1) The reactor is at < 5% power;

Table 15.4-11

MSLBA Parameters

I~~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~~ 7,400 (0.2) ~~μCi/g DE I-131~~
C.	Steam Mass Released, kg (lbm)	~~4,705~~20,369 (44905)
D.	Water Mass Released, kg (lbm)	~~82,328~~43,184 (95204)
~~2~~II. Data and assumptions used to estimate activity released		
A.	Isolation valve closure time, sec	5
B.	MSIV Response time, sec	0.5
C.	Total assumed release duration, sec	5.5
III. Control Room Parameters		
A.	Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E4)
B.	Unfiltered intake, l/s (cfm)	200 (424)
C.	Filtered intake, l/s (cfm)	0 (0)
D.	Unfiltered inleakage, l/s (cfm)	0 (0)
E.	Occupancy Factors	
	0 – 1 day	1.0
	1 – 4 days	0.6
	4 – 30 days	0.4

~~3~~IV Dispersion Data .		
A.	Off-site Meteorology	~~2.00E-03 s/m^3~~
	Exclusion Area Boundary	
	0 – 2 hrs	2.00E-03 s/m^3
	> 2 hrs	NR*
	Low Population	

0 – 8 hrs	1.90E-04 s/m^3
> 8 hrs	NR*
B. Control Room Meteorology (Turbine Building Release Point)	
0 – 2 hrs	1.20E-03 s/m^3
> 2 hrs	NR*
~~B~~C. Method of Dose Calculation	RG 1.183
D~~C~~. Dose Conversion Assumptions	RG 1.183
E~~D~~. Activity Inventory and Releases	Tables 15.4-12
F~~E~~. Dose Evaluations	Table 15.4-13

Note * Due to the short release, values > 2 hours do not impact the calculated doses, therefore they are Not Required.

Table 15.4-12
MSLBA Environment Releases

Isotope	Equilibrium Iodine		Iodine Spike	
	MBq	Ci	MBq	Ci
Co-58	7.2E+02	1.9E-02	7.2E+02	1.9E-02
Co-60	1.4E+03	3.8E-02	1.4E+03	3.8E-02
Kr-85	7.4E+00	2.0E-04	7.4E+00	2.0E-04
Kr-85m	1.9E+03	5.1E-02	1.9E+03	5.1E-02
Kr-87	6.1E+03	1.6E-01	6.1E+03	1.6E-01
Kr-88	6.1E+03	1.6E-01	6.1E+03	1.6E-01
Rb-86	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sr-89	7.2E+02	1.9E-02	7.2E+02	1.9E-02
Sr-90	5.0E+01	1.3E-03	5.0E+01	1.3E-03
Sr-91	2.7E+04	7.3E-01	2.7E+04	7.3E-01
Sr-92	6.6E+04	1.8E+00	6.6E+04	1.8E+00
Y-90	5.0E+01	1.3E-03	5.0E+01	1.3E-03
Y-91	2.9E+02	7.8E-03	2.9E+02	7.8E-03
Y-92	4.0E+04	1.1E+00	4.0E+04	1.1E+00
Y-93	2.7E+04	7.3E-01	2.7E+04	7.3E-01
Zr-95	5.8E+01	1.6E-03	5.8E+01	1.6E-03
Zr-97	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Nb-95	5.8E+01	1.6E-03	5.8E+01	1.6E-03
Mo-99	1.4E+04	3.8E-01	1.4E+04	3.8E-01
Tc-99m	1.4E+04	3.8E-01	1.4E+04	3.8E-01
Ru-103	1.4E+02	3.8E-03	1.4E+02	3.8E-03
Ru-105	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ru-106	2.1E+01	5.6E-04	2.1E+01	5.6E-04
Rh-105	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sb-127	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Sb-129	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-127	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-127m	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-129	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te-129m	2.9E+02	7.8E-03	2.9E+02	7.8E-03
Te-131m	7.0E+02	1.9E-02	7.0E+02	1.9E-02
Te-132	7.2E+01	1.9E-03	7.2E+01	1.9E-03
I-131	1.3E+05	3.5E+00	2.6E+06	7.0E+01
I-132	1.2E+06	3.3E+01	2.4E+07	6.6E+02

Isotope	Equilibrium Iodine		Iodine Spike	
	MBq	Ci	MBq	Ci
I-133	8.9E+05	2.4E+01	1.8E+07	4.8E+02
I-134	2.2E+06	6.1E+01	4.5E+07	1.2E+03
I-135	1.3E+06	3.4E+01	2.5E+07	6.8E+02
Xe-133	2.6E+03	6.9E-02	2.6E+03	6.9E-02
Xe-135	7.1E+03	1.9E-01	7.1E+03	1.9E-01
Cs-134	1.9E+02	5.2E-03	1.9E+02	5.2E-03
Cs-136	1.3E+02	3.5E-03	1.3E+02	3.5E-03
Cs-137	5.1E+02	1.4E-02	5.1E+02	1.4E-02
Ba-139	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ba-140	2.9E+03	7.8E-02	2.9E+03	7.8E-02
La-140	2.9E+03	7.8E-02	2.9E+03	7.8E-02
La-141	0.0E+00	0.0E+00	0.0E+00	0.0E+00
La-142	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-141	2.1E+02	5.6E-03	2.1E+02	5.6E-03
Ce-143	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-144	2.1E+01	5.6E-04	2.1E+01	5.6E-04
Pr-143	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Nd-147	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Np-239	5.8E+04	1.6E+00	5.8E+04	1.6E+00
Pu-238	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pu-239	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pu-240	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pu-241	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Am-241	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cm-242	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cm-244	0.0E+00	0.0E+00	0.0E+00	0.0E+00

~~Isotope~~	~~Equilibrium Activity MBq~~	~~Iodine Spike Activity MBq~~	~~Isotope~~	~~Equilibrium Activity MBq~~	~~Iodine Spike Activity MBq~~
~~Co-58~~	~~1.4E+03~~	~~1.4E+03~~	~~Te-131m~~	~~1.3E+03~~	~~1.3E+03~~
~~Co-60~~	~~2.7E+03~~	~~2.7E+03~~	~~Te-132~~	~~1.4E+02~~	~~1.4E+02~~
~~Kr-85~~	~~1.7E+00~~	~~1.7E+00~~	~~I-131~~	~~2.4E+05~~	~~4.9E+06~~
~~Kr-85m~~	~~4.4E+02~~	~~4.4E+02~~	~~I-132~~	~~2.3E+06~~	~~4.6E+07~~
~~Kr-87~~	~~1.4E+03~~	~~1.4E+03~~	~~I-133~~	~~1.7E+06~~	~~3.4E+07~~
~~Kr-88~~	~~1.4E+03~~	~~1.4E+03~~	~~I-134~~	~~4.2E+06~~	~~8.5E+07~~
~~Rb-86~~	~~0.0E+00~~	~~0.0E+00~~	~~I-135~~	~~2.4E+06~~	~~4.7E+07~~
~~Sr-89~~	~~1.4E+03~~	~~1.4E+03~~	~~Xe-133~~	~~5.9E+02~~	~~5.9E+02~~
~~Sr-90~~	~~9.4E+01~~	~~9.4E+01~~	~~Xe-135~~	~~1.6E+03~~	~~1.6E+03~~
~~Sr-91~~	~~5.2E+04~~	~~5.2E+04~~	~~Cs-134~~	~~3.7E+02~~	~~3.7E+02~~

~~Sr-92~~	~~1.2E+05~~	~~1.2E+05~~	~~Cs-136~~	~~2.4E+02~~	~~2.4E+02~~
~~Y-90~~	~~9.4E+01~~	~~9.4E+01~~	~~Cs-137~~	~~9.7E+02~~	~~9.7E+02~~
~~Y-91~~	~~5.5E+02~~	~~5.5E+02~~	~~Ba-139~~	~~0.0E+00~~	~~0.0E+00~~
~~Y-92~~	~~7.6E+04~~	~~7.6E+04~~	~~Ba-140~~	~~5.5E+03~~	~~5.5E+03~~
~~Y-93~~	~~5.2E+04~~	~~5.2E+04~~	~~La-140~~	~~5.5E+03~~	~~5.5E+03~~
~~Zr-95~~	~~1.1E+02~~	~~1.1E+02~~	~~La-141~~	~~0.0E+00~~	~~0.0E+00~~
~~Zr-97~~	~~0.0E+00~~	~~0.0E+00~~	~~La-142~~	~~0.0E+00~~	~~0.0E+00~~
~~Nb-95~~	~~1.1E+02~~	~~1.1E+02~~	~~Ce-141~~	~~4.0E+02~~	~~4.0E+02~~
~~Mo-99~~	~~2.7E+04~~	~~2.7E+04~~	~~Ce-143~~	~~0.0E+00~~	~~0.0E+00~~
~~Tc-99m~~	~~2.7E+04~~	~~2.7E+04~~	~~Ce-144~~	~~4.0E+01~~	~~4.0E+01~~
~~Ru-103~~	~~2.7E+02~~	~~2.7E+02~~	~~Pr-143~~	~~0.0E+00~~	~~0.0E+00~~
~~Ru-105~~	~~0.0E+00~~	~~0.0E+00~~	~~Nd-147~~	~~0.0E+00~~	~~0.0E+00~~
~~Ru-106~~	~~4.0E+01~~	~~4.0E+01~~	~~Np-239~~	~~1.1E+05~~	~~1.1E+05~~
~~Rh-105~~	~~0.0E+00~~	~~0.0E+00~~	~~Pu-238~~	~~0.0E+00~~	~~0.0E+00~~
~~Sb-127~~	~~0.0E+00~~	~~0.0E+00~~	~~Pu-239~~	~~0.0E+00~~	~~0.0E+00~~
~~Sb-129~~	~~0.0E+00~~	~~0.0E+00~~	~~Pu-240~~	~~0.0E+00~~	~~0.0E+00~~
~~Te-127~~	~~0.0E+00~~	~~0.0E+00~~	~~Pu-241~~	~~0.0E+00~~	~~0.0E+00~~
~~Te-127m~~	~~0.0E+00~~	~~0.0E+00~~	~~Am-241~~	~~0.0E+00~~	~~0.0E+00~~
~~Te-129~~	~~0.0E+00~~	~~0.0E+00~~	~~Cm-242~~	~~0.0E+00~~	~~0.0E+00~~
~~Te-129m~~	~~5.5E+02~~	~~5.5E+02~~	~~Cm-244~~	~~0.0E+00~~	~~0.0E+00~~

Table 15.4-13

MSLBA 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~~ any (worst) 2 hour period		
Pre-incident Spike	~~12.6~~6.7	25
Equilibrium Iodine Activity	~~0.7~~0.4	2.5
Outer Boundary of Low Population Zone (LPZ) for the ~~Entire Period of the Radioactive Cloud Passage~~Duration of the Accident (30 days)		
Pre-incident Spike	~~12.6~~<0.1	25
Equilibrium Iodine Activity	~~0.7~~0.6	2.5
Control Room Dose for the Duration of the Accident (30 days) ~~4.5~~ ~~5~~		
Pre-incident Spike	0.13	5
Equilibrium Iodine Activity	2.41	5