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Proprietary Notice
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 07-100

Docket No. 52-010

March 26, 2007

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

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application Safety Analysis – RAI Numbers 15.4-2 and 15.4-3**

Enclosure 1 contains proprietary information as defined in 10CFR2.390. The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GE. GE hereby requests that the proprietary information in Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17. A non proprietary version is contained in Enclosure 2. Enclosure 4 is a proprietary CD labeled RADTRAD v3.0.3 Input and Output Deck Files – GE Proprietary for the Main Steam Line Break and Instrument Line Break dose consequences analyses.

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

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Reference:

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

Enclosures:

1. MFN 07-100 – Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Numbers 15.4-2 and 15.4-3 – GE Proprietary Information
2. MFN 07-100 – Response to Portion of NRC Request for Additional Information Letter No. 69 - Safety Analysis – RAI Numbers 15.4-2 and 15.4-3 – Non Proprietary Version
3. Affidavit – David H. Hinds – dated March 26, 2007
4. MFN 07-100 – Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RADTRAD v3.0.3 Input and Output Deck Files – GE Proprietary
    a. ESBWR MSLB Rev 1TSMax.psf ("Iodine Spike" input file)
    b. ESBWR MSLB Rev 1TSMax.o0("Iodine Spike" output file)
    c. ESBWR MSLB Rev1 TSMax.nif ("Iodine Spike" nuclide inventory file)
    d. ESBWR MSLB Rev1 TSEq.psf ("Equilibrium iodine" input file)
    e. ESBWR MSLB Rev1 TSEq.o0 ("Equilibrium iodine" output file)
    f. ESBWR MSLB Rev1 TSEq.nif ("Equilibrium Spike" nuclide inventory file)
    g. ESBWR MSLB Rev1.rft (Release fraction file – common to both)
    h. ESBWR ILB Rev1 TSMax.nif
    i. ESBWR ILB Rev1.psf
    j. ESBWR ILB Rev1.o0
    k. ESBWR ILB Rev1.rft

cc: AE Cubbage USNRC (with enclosures)
GB Stramback GE/San Jose (with enclosures)
eDRF 0064-2601

Enclosure 2

MFN 07-100

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

Safety Analysis

RAI Numbers 15.4-2, 15.4-3

Non-Proprietary Version

NRC RAI 15.4-2:

Question Summary: Provide source term assumptions for main steamline break accident Outside Containment

Full Text: DCD Tier 2, Revision 1, Section 15.4.5, "Main Steamline Break Accident Outside Containment," describes the postulated a large steam line break outside containment 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 state if the five second main steam isolation valve closure time, assumed in the radiological analysis, is specified in the ESBWR Technical Specification.*
>
> *(B) Please provide control room operator doses for both with an assumed pre-accident iodine spike and an accident-initiated iodine spike.*
>
> *(C) Please provide complete 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.*

GE Response:

General Information Concerning the MSLB: The dose consequence analysis of a Main Steam Line Break Outside of Containment (MSLB) documented in DCD Subsection 15.4.5, Revision 2, to address several issues:

- **Mass Releases:** The mass release assumed in the DCD Tier 2, Subsection 15.4.5 was revised. The mass release assumed in DCD Revision 2 represented the mass release from the Lungmen ABWR with a safety factor of 1.5 applied. Additional review by GE concluded that this safety factor of 1.5 was over conservative. The DCD, Revision 3 analysis applied a safety factor of 1.25. GE is preparing a formal MSLB outside of containment analysis to determine exact mass release values.
- **Atmospheric Dispersion Factors:** The previous analysis did not take into account site-specific information in determining atmospheric dispersion factors. GE has performed additional research to determine conservative X/Q values for the ESBWR.
- **Control Room Ventilation Model:** The DCD Revision 3 MSLB dose consequence analysis assumes that control room ventilation operates in normal mode for the duration of the event. No credit is taken for control room filter trains.

NRC Item (A): *Please state if the five second main steam isolation valve closure time, assumed in the radiological analysis, is specified in the ESBWR Technical Specification.*

GE Response to Item (A):

DCD, Chapter 16, Technical Specifications (TS) Surveillance Requirements (SR) 3.6.1.3.5 states "Verify the full closure isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds."

NRC Item (B): *Please provide control room operator doses for both with an assumed pre-accident iodine spike and an accident-initiated iodine spike.*

GE Response to Item (B):

GE has determined doses for control room operators for the accident-initiated iodine spike, which bounds the pre-accident iodine spike. The results are included in the DCD mark-ups included in this transmittal.

NRC Item (C): *Please provide complete 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.*

GE Response to Item (C):

The NRC computer code RADTRAD v3.0.3 was used to evaluate the dose consequences from a MSLB. [[

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Affected Documents:

Changes to DCD Tier 2 Tables 15.4-10, 15.4-11, 15.4-12 and 15.4-13 were implemented in Revision 3 of Chapter 15.

NRC RAI 15.4-3:

Question Summary: Provide source

Full Text: DCD Tier 2, Revision 1, Section 15.4.8, "Failure of Small Line Carrying Primary Coolant -23-term assumptions for Failure of Small Line Carrying Primary Coolant Outside Containment Outside Containment," describes the postulated a small steam or liquid line break inside or outside the containment 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) You stated in DCD Tier 2, Revision 1, Section 15.4.8.5.1 that the SRP does not provide detailed guidance. The staff believes the detailed guidance is provided in SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment" Revision 2, July 1981. Please state if you have taken any exceptions or deviations from the guidance provided in SRP Section 15.6.2.

(B) Please provide steam/water break flow rate(s) used in your dose calculation.

(C) Please provide a copy of dose calculation performed including determination of iodine appearance rates and resulting iodine concentrations due to the iodine spike.

(D) You used reactor building flow (leak) rate of 200 percent per hour. Is this value in the ESBWR technical specification? 10 CFR 50.36(c)(ii)(C) criterions requires that a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier should be specified as a limiting condition for operation.

GE Response:

General Information Concerning the Small Line Break Outside of Containment: The dose consequence analysis of a Small Line Break Outside of Containment, or commonly referred to as an Instrument Line Break (ILB) is documented in DCD Subsection 15.4.5. The DCD Revision 2 analysis was revised to be consistent with the Alternative Source Term dose methodology consistent with NUREG-1465 and Regulatory Guide 1.183. Other changes were required to ensure the event is consistent with the revised Loss of Coolant Accident dose consequence analysis, which was previously submitted to the NRC via NEDE-33279P. A summary of the changes to the ILB analysis follows.

- **Control Room Doses:** The analysis documented in DCD Tier 2, Section 15.4.5, Revision 2 did not document doses to control room operators. The main control room dose acceptance criteria is based on 10 CFR 50, Appendix A, General Design Criteria 19, which provides a dose limit of 5 REM TEDE for control room operators. The control room envelope is conservatively assumed to operate in normal mode for the duration of the event. The control room doses do not credit operation of the charcoal filter trains. The results for control room operators were calculated as indicated in DCD Tier 2, Revision 3.

- **Computer Code:** The DCD Tier 2, Subsection 15.4.5, Revision 2 dose consequence for "whole body" and "thyroid" utilized the GE code CONAC04A. TEDE doses were estimated using the CONAC04A results. The revised analysis documented in DCD Tier 2, Revision 3, utilizes the NRC computer code RADTRAD (v3.0.3), which directly calculates dose consequences in terms of TEDE.
- **Atmospheric Dispersion Factors:** The DCD Tier 2, Revision 3 analysis did not take into account site-specific information in determining atmospheric dispersion factors. GE has performed additional research to determine conservative X/Q values for the ESBWR.
- **Reactor Building Release Rate:** The release rate from the Reactor Building was assumed to be 200% per day for the DCD Tier 2, Subsection 15.4.5, Revision 2 dose consequence analysis. The revised LOCA dose analysis (NEDE-33279P) assumed a leakage rate of 50% per day. The revised ILB analysis conservatively assumes 10x that value, or a total of 500% per day.

The DCD is updated accordingly as indicated on the attached markups. The responses to the NRC questions account for the revised DCD.

<u>NRC Item (A):</u> *You stated in DCD Tier 2, Revision 1, Section 15.4.8.5.1 that the SRP does not provide detailed guidance. The staff believes the detailed guidance is provided in SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, July 1981. Please state if you have taken any exceptions or deviations from the guidance provided in SRP Section 15.6.2.*

<u>GE Response to Item (A)</u>:

GE agrees that the guidance in SRP 15.6.2 is applicable to small line breaks. The SRP refers to 10 CFR 100.11, whereas the ESBWR utilizes the dose consequence acceptance criteria set forth in 10 CFR 50.34(a)(1). As such, the revised (DCD, Revision 3) analysis inherently uses the Alternative Source Term dose methodology in accordance with NUREG-1465 and Regulatory Guide 1.183. The acceptance criteria used by GE was 10% of 10 CFR 50.34(a)(1) for off-site doses (2.5 REM TEDE), and 10 CFR 50, Appendix A, GDC 19 for control room doses (5.0 REM TEDE).

GE used the guidance provided in SRP 15.6.2, DRAFT Revision 3 and SRP 15.6.2, Revision 2 as guidance in preparing the revised analysis. The only notable exceptions are the dose acceptance criteria previously discussed, and the "iodine spike" concentration. GE utilized the guidance found in Regulatory Guide 1.183, Appendix D, Section 2.1 where the "iodine spike" concentration is based on TS 3.4.3, or 148000 Bq/g (4.0 μCi/g) rather than 500x the "equilibrium" value.

NRC Item (B): Please provide steam/water break flow rate(s) used in your dose calculation.

GE Response to Item (B):

The coolant (water) flow rate assumed for the ILB analysis is based on information from GE's CONAC04A code, as documented in NEDO-32708, "Radiological Accident Evaluation: The CONAC04A Code," August 1997. [[

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NRC Item (C): *Please provide a copy of dose calculation performed including determination of iodine appearance rates and resulting iodine concentrations due to the iodine spike.*

GE Response to Item (C):

The NRC computer code RADTRAD v 3.0.3 was used to calculate the dose consequences from an ILB. [[

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NRC Item (D): *You used reactor building flow (leak) rate of 200 percent per hour. Is this value in the ESBWR technical specification? 10 CFR 50.36(c)(ii)(C) criterions requires that a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier should be specified as a limiting condition for operation.*

GE Response to Item (D):

Chapter 16 of DCD, Tier 2 contains the plant Technical Specifications. Surveillance Requirement 3.6.3.1.4 requires licensees to "Verify Reactor Building exfiltration rate [are] within limits." The LOCA dose calculation documented in NEDE-33279P assumed a flow rate of 50% per day from the Reactor Building. The revised ILB analysis conservatively assumes 10x the value assumed for the LOCA calculation, or 500% per day.

Affected Documents:

Changes to DCD Tier 2 Subsection 15.4.5.5.1, Subsection 15.4.8.5.1, Table 15.4-17 and Table 15.4-18 were implemented in Revision 3 of Chapter 15.

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:

(i) 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).

(ii) 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).

(iii) 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 following are assumptions in the radiological analysis that require confirmation by the COL applicant:

- The atmospheric dispersion parameters are as stated in Table 15.4-5.

15.4.4.6.2 Assumptions to be Confirmed by the COL Holder

The following are assumptions in the radiological analysis that require confirmation by the COL holder:

- The Reactor Building leakage rate is assumed to be 50 volume % per day.
- The unfiltered inleakage term assumed in the analysis bound those of the actual control room.

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 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 from the Turbine Building of 1.0E+08 volume % per day. 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.

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 Class 1E 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;

(2) there are failures of both Class 1E separation-detection devices or a failure of the rod block interlock;

(3) there is a failure of the latch mechanism to occur;

(4) simultaneously, there is a additional failure that causes the occurrence of a stuck rod on the same FMCRD;

15.4.7.5.2 Fission Product Release

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 iodine concentration assumed is that of the maximum equilibrium reactor water concentration used for the MSLBA, subject to a 2% carryover of iodine in the water to steam condensate. Noble gas activity in the condensate is negligible, because the air ejectors remove all noble gases from the condenser.

15.4.7.5.3 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.

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

15.4.7.5.4 Assumptions to be Confirmed by the COL Applicant

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 Results

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

15.4.8 Failure of Small Line Carrying Primary Coolant Outside Containment

This event postulates a small steam or liquid line pipe break inside or outside the containment, but within a controlled release structure. To bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where detection is not automatic or apparent. This event is less limiting from a core performance evaluation standpoint than the postulated events presented in Subsections 15.4.5 (Main Steamline Break Accident Outside Containment), 15.4.4 (Loss-of-Coolant Accident Inside Containment Radiological Analysis), and 15.4.7 (Feedwater Line Break Outside Containment).

This postulated event represents the envelope evaluation for small line failure inside and outside the containment relative to sensitivity for detection.

15.4.8.1 Identification of Causes

There is no identified specific event or circumstance that results in the failure of an instrument line. These lines are designed to high quality, engineering standards, seismic and environmental requirements. They also are equipped with either excess flow check valves or isolation valves.

However, for the purpose of evaluating the consequences of a small line rupture, the rupture of an instrument line is assumed to occur along with a failure to isolate the break.

A circumferential rupture of an instrument line that is connected to the primary coolant system is postulated to occur outside the drywell, but inside the Reactor Building. The associated effects from a rupture in the drywell would not be as significant as those from the failure in the Reactor Building.

15.4.8.2 Sequence of Events and Systems Operations

15.4.8.2.1 Sequence of Events

The leak may result in noticeable increases in radiation, temperature, humidity, or audible noise levels in the Reactor Building or abnormal indications of actuations caused by the affected instrument.

Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolatable leak. The action consists of the orderly shutdown and depressurization of the reactor.

15.4.8.2.2 Systems Operation

A presentation of plant, RPS, ESF and other safety-related action are given in Sections 6.3, 7.3 and 7.6.

15.4.8.2.3 The Effect of Single Failures and Operator Errors

There is no single failure or operator error that significantly affects the system response to this event. Single failures in other instrument channels could lead to actuation of ESF actions such as reactor scram or MSIV closure under the assumption that the line break trips one division and the single failure trips another division to produce a 2-out-of-4 trip condition.

15.4.8.3 Core and System Performance

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.4.4, 15.4.5 and 15.4.7. Consequently, instrument line breaks are considered to be bounded specifically by the MSLBA (Subsection 15.4.5). Details of this calculation, including those pertinent to core and system performance, are presented in Subsection 15.4.5.3.

15.4.8.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 15.4-17.

15.4.8.3.2 Results

No fuel damage or core uncovery occurs as a result of this accident. Instrument line breaks are within the spectrum considered in ECCS performance calculations presented in Section 6.3.

15.4.8.4 Barrier Performance

The following assumptions and conditions are the basis for the mass loss during the release period of this event.

- The instrument line releases coolant into the Reactor Building for 30 minutes at normal operating temperature and pressure. Following this time period, the operator detects the event, scrams the reactor and initiates reactor depressurization.
- Reactor coolant is released to the Reactor Building, until the reactor is depressurized.
- The flow from the instrument line is limited by reactor pressure and a 6-mm (0.25-inch) diameter flow restricting orifice inside the drywell. The Moody critical blowdown model is applicable, and the flow is critical at the orifice (Reference 15.4-6).

15.4.8.5 Radiological Analysis

15.4.8.5.1 General

The radiological analysis is based upon conservative assumptions considered acceptable to the NRC. The assumptions found in Table 15.4-17 assume that all of the iodine available in the flashed water is transported from the Reactor Building to the environment without treatment.

15.4.8.5.2 Fission Product Release

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 amount of iodine released to the Reactor Building atmosphere and to the environment is presented in Table 15.4-18.

15.4.8.5.3 Results

The calculated exposures for the analysis are presented in Table 15.4-19, and are less than the regulatory guideline exposures.

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 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.

15.4.11 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:
  - 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 Steam Line Break Accident:
  - Releases from the Turbine Building must have control room air intake X/Q values less than or equal to those presented in Table 2.0-1.
- Instrument Line Break Accident:
  - Releases from the Reactor Building must have control room air intake X/Q values less than or equal to those presented in Table 2.0-1.
- 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.

15.4-4 GE Nuclear Energy, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," NEDC-31858P (GE proprietary), Revision 2, September 1993.

15.4-5 General Electric Company, "Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes," 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

15.4-6 General Electric Company, "Maximum Two-Phase Vessel Blowdown from Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

Table 15.4-10

Sequence of Events for Main Steamline Break Accident (MSLBA) Outside Containment

Time (sec)	**Event**
0	Guillotine break of one main steam line outside containment.
0.5	High steamline flow signal initiates closure of MSIVs
< 1.0	Reactor begins scram.
< 2.0	Partial closure (15%) of MSIV initiates isolation condensers.
< 5.5	MSIVs fully closed.
10	Reactor low water Level 2 is reached. Isolation condensers receive second initiation signal.
32	Isolation condensers in full operation. Water level stabilized.
435	SRVs open on high vessel pressure (if isolation condensers are not available). The SRVs open and close to maintain vessel absolute pressure.
3540	Reactor low water Level 1 is reached (if isolation condensers are not available). ADS timer initiated.
3550	ADS timer timed out. ADS actuation sequence initiated. GDCS timer initiated.
3700	GDCS timer timed out. GDCS injection valves open.
3880	Vessel pressure decreases below shutoff head of GDCS. GDCS reflooding flow into the vessel begins.
* The core remains covered throughout the transient and no core heatup occurs.	

Table 15.4-11

MSLBA Parameters

1.	Data and assumptions used to estimate source terms	
	A. Fuel Damage	none
	B. Reactor Coolant Activity: Pre-incident Spike Equilibrium Iodine Activity	 4.0 μCi/g DE I-131 0.2 μCi/g DE I-131
	C. Steam Mass Released, kg	4,705
	D. Water Mass Released, kg	82,328
2.	Data and assumptions used to estimate activity released	
	A. Isolation valve closure time, sec	5
	B. MSIV Response time, sec	0.5
3.	Dispersion Data	
	A. Off-site Meteorology	2.00E-03 s/m^3
	B. Method of Dose Calculation	RG 1.183
	C. Dose Conversion Assumptions	RG 1.183
	D. Activity Inventory and Releases	Tables 15.4-12
	E. Dose Evaluations	Table 15.4-13

Table 15.4-12
MSLBA Environment Releases

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		
Pre-incident Spike	12.6	25
Equilibrium Iodine Activity	0.7	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage		
Pre-incident Spike	12.6	25
Equilibrium Iodine Activity	0.7	2.5
Control Room Dose for the Duration of the Accident	4.5	5

Table 15.4-17

Instrument Line Break Accident Parameters

I. Data and assumptions used to estimate source terms	
A. Power level, MWt	4590
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 accident, hr	6
E. Number of bundles in core	1132
II. Data and assumptions used to estimate activity released	
A. Iodine water concentration	4.0 μCi/g DE I-131
B. Iodine plateout fraction, %	0
C. Reactor Building Flow rate, %/hour	500
III Dispersion and Dose Data	
A. Meteorology	Table 15.4-9 (Reactor Building)
B. Method of Dose Calculation	Ref 15.4-1
C. Dose conversion Assumptions	RG 1.183 and Ref. 15.4-1
D. Activity Inventory/releases	Table 15.4-18
E. Dose evaluations	Table 15.4-19

Table 15.4-18
Instrument Line Break Accident Isotopic Inventory (MBq)

Time (hr)	0.02	0.17	0.5	1	2	4	8	12
Co-58	1.2E-03	8.7E-02	7.4E-01	2.6E+00	8.2E+00	2.4E+01	4.8E+01	5.9E+01
Co-60	2.4E-03	1.7E-01	1.5E+00	5.1E+00	1.6E+01	4.8E+01	9.6E+01	1.2E+02
Rb-86	5.2E-01	3.7E+01	3.1E+02	1.1E+03	3.5E+03	1.0E+04	2.0E+04	2.5E+04
Sr-89	1.2E-03	8.7E-02	7.4E-01	2.6E+00	8.2E+00	2.4E+01	4.8E+01	5.9E+01
Sr-90	8.4E-05	6.0E-03	5.1E-02	1.8E-01	5.6E-01	1.7E+00	3.3E+00	4.0E+00
Sr-91	4.6E-02	3.3E+00	2.8E+01	9.7E+01	3.1E+02	9.2E+02	1.8E+03	2.2E+03
Sr-92	1.1E-01	7.9E+00	6.7E+01	2.3E+02	7.5E+02	2.2E+03	4.4E+03	5.3E+03
Y-90	8.4E-05	6.0E-03	5.1E-02	1.8E-01	5.6E-01	1.7E+00	3.3E+00	4.0E+00
Y-91	4.9E-04	3.5E-02	3.0E-01	1.0E+00	3.3E+00	9.7E+00	1.9E+01	2.3E+01
Y-92	6.8E-02	4.8E+00	4.1E+01	1.4E+02	4.6E+02	1.4E+03	2.7E+03	3.3E+03
Y-93	4.6E-02	3.3E+00	2.8E+01	9.7E+01	3.1E+02	9.2E+02	1.8E+03	2.2E+03
Zr-95	9.8E-05	7.0E-03	5.9E-02	2.0E-01	6.6E-01	1.9E+00	3.9E+00	4.7E+00
Nb-95	9.8E-05	7.0E-03	5.9E-02	2.0E-01	6.6E-01	1.9E+00	3.9E+00	4.7E+00
Mo-99	2.4E-02	1.7E+00	1.5E+01	5.1E+01	1.6E+02	4.8E+02	9.6E+02	1.2E+03
Tc-99m	2.4E-02	1.7E+00	1.5E+01	5.1E+01	1.6E+02	4.8E+02	9.6E+02	1.2E+03
Ru-103	2.4E-04	1.7E-02	1.5E-01	5.1E-01	1.6E+00	4.8E+00	9.6E+00	1.2E+01
Ru-106	3.5E-05	2.5E-03	2.1E-02	7.4E-02	2.4E-01	7.0E-01	1.4E+00	1.7E+00
Te-129m	4.9E-04	3.5E-02	3.0E-01	1.0E+00	3.3E+00	9.7E+00	1.9E+01	2.3E+01
Te-131m	1.2E-03	8.5E-02	7.2E-01	2.5E+00	8.0E+00	2.4E+01	4.7E+01	5.7E+01
Te-132	1.2E-04	8.7E-03	7.4E-02	2.6E-01	8.2E-01	2.4E+00	4.8E+00	5.9E+00
I-131	4.3E+00	3.1E+02	2.6E+03	9.1E+03	2.9E+04	8.6E+04	1.7E+05	2.1E+05
I-132	4.1E+01	2.9E+03	2.5E+04	8.6E+04	2.8E+05	8.2E+05	1.6E+06	2.0E+06
I-133	3.0E+01	2.1E+03	1.8E+04	6.3E+04	2.0E+05	6.0E+05	1.2E+06	1.4E+06
I-134	7.5E+01	5.4E+03	4.5E+04	1.6E+05	5.1E+05	1.5E+06	3.0E+06	3.6E+06
I-135	4.2E+01	3.0E+03	2.6E+04	8.8E+04	2.8E+05	8.4E+05	1.7E+06	2.0E+06
Cs-134	3.3E-04	2.3E-02	2.0E-01	6.8E-01	2.2E+00	6.5E+00	1.3E+01	1.6E+01
Cs-136	2.2E-04	1.6E-02	1.3E-01	4.6E-01	1.5E+00	4.3E+00	8.6E+00	1.0E+01
Cs-137	8.7E-04	6.2E-02	5.2E-01	1.8E+00	5.8E+00	1.7E+01	3.4E+01	4.2E+01
Ba-140	4.9E-03	3.5E-01	3.0E+00	1.0E+01	3.3E+01	9.7E+01	1.9E+02	2.3E+02
La-140	4.9E-03	3.5E-01	3.0E+00	1.0E+01	3.3E+01	9.7E+01	1.9E+02	2.3E+02
Ce-141	3.5E-04	2.5E-02	2.1E-01	7.4E-01	2.4E+00	7.0E+00	1.4E+01	1.7E+01
Ce-144	3.5E-05	2.5E-03	2.1E-02	7.4E-02	2.4E-01	7.0E-01	1.4E+00	1.7E+00
Np-239	9.8E-02	7.0E+00	5.9E+01	2.0E+02	6.6E+02	1.9E+03	3.9E+03	4.7E+03

Table 15.4-19

Instrument Line Break Accident 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	0.15	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage*		2.5
8 hours (0 – 8 hr X/Q = 1.90E-04 sec/m^3)	0.04	
1 day (8 – 24 hr X/Q = 1.40E-04 sec/m^3)	0.05	
4 days (24 – 96 hr X/Q = 7.50E-05 sec/m^3)	0.05	
30 days (96 – 720 hr X/Q = 3.00E-05 sec/m^3)	0.05	
Main Control Room	0.2	5.0

Note *: Cumulative dose up to stated time period.

Enclosure 3

MFN 07-100

Affidavit

General Electric Company

AFFIDAVIT

I, David H. Hinds, state as follows:

(1) I am Manager, New Projects Engineering, General Electric Company ("GE") and 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 1 (CD) of GE letter MFN 07-100, Mr. James C. Kinsey to U.S. Nuclear Regulatory Commission, *MFN 07-100 – Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Numbers 15.4-2 and 15.4-3* dated March 26, 2007. The proprietary information is in Enclosures 1 and 4, respectively, *MFN 07-100 – Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Numbers 15.4-2 and 15.4-3 – GE Proprietary Information, and Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RADTRAD v3.0.3 Input and Output Deck Files – GE Proprietary:*

a. *ESBWR MSLB Rev 1TSMax.psf ("Iodine Spike" input file)*
b. *ESBWR MSLB Rev 1TSMax.o0("Iodine Spike" output file)*
c. *ESBWR MSLB Rev1 TSMax.nif ("Iodine Spike" nuclide inventory file)*
d. *ESBWR MSLB Rev1 TSEq.psf ("Equilibrium iodine" input file)*
e. *ESBWR MSLB Rev1 TSEq.o0 ("Equilibrium iodine" output file)*
f. *ESBWR MSLB Rev1 TSEq.nif ("Equilibrium Spike" nuclide inventory file)*
g. *ESBWR MSLB Rev1.rft (Release fraction file – common to both)*
h. *ESBWR ILB Rev1 TSMax.nif*
i. *ESBWR ILB Rev1.psf*
j. *ESBWR ILB Rev1.o0*
k. *ESBWR ILB Rev1.rft*

Enclosure 1 and Enclosure 4 CD label contains the designation "GE Proprietary {3}." The superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(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 2.790(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, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 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 General Electric's competitors without license from General Electric 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 General Electric customer-funded development plans and programs, resulting in potential products to General Electric;
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, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, 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 describes the models and methodologies GE will use in evaluating the dose consequences of design basis accidents (DBAs) for the ESBWR. GE and its partners performed significant additional research and evaluation to develop a basis for these revised methodologies to be used in evaluating the ESBWR over a period of several years at a cost of over one million dollars.

(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 26th day of March 2007.

David H. Hinds
General Electric Company

Enclosure 4

MFN 07-100

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

Safety Analysis

CD - RADTRAD v3.0.3

Input and Output Deck Files – GE Proprietary

a. ESBWR MSLB Rev 1TSMax.psf ("Iodine Spike" input file)
b. ESBWR MSLB Rev 1TSMax.o0("Iodine Spike" output file)
c. ESBWR MSLB Rev1 TSMax.nif ("Iodine Spike" nuclide inventory file)
d. ESBWR MSLB Rev1 TSEq.psf ("Equilibrium iodine" input file)
e. ESBWR MSLB Rev1 TSEq.o0 ("Equilibrium iodine" output file)
f. ESBWR MSLB Rev1 TSEq.nif ("Equilibrium Spike" nuclide inventory file)
g. ESBWR MSLB Rev1.rft (Release fraction file – common to both)
h. ESBWR ILB Rev1 TSMax.nif
i. ESBWR ILB Rev1.psf
j. ESBWR ILB Rev1.o0
k. ESBWR ILB Rev1.rft