FPL Energy
Seabrook Station

FPL Energy Seabrook Station
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October 6, 2003

Docket No. 50-443

NYN-03061

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Seabrook Station
License Amendment Request 03-02
"Implementation of Alternate Source Term"

FPL Energy Seabrook, LLC (FPLE Seabrook) is providing within Enclosure 1, License Amendment Request (LAR) 03-02. LAR 03-02 is submitted pursuant to the requirements of 10 CFR 50.4, 10 CFR 50.67, and 10 CFR 50.90. FPLE Seabrook requests approval of full implementation of an alternate source term (AST) for Seabrook Station. This LAR also proposes to revise the Technical Specification (TS) definition of "Dose Equivalent I-131."

As discussed in Section IV of the enclosed, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92. A copy of this letter and the enclosed LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b). FPLE Seabrook requests NRC Staff review of LAR 03-02 and issuance of a license amendment by September 25, 2004 (see Section V of Enclosure 1).

FPLE Seabrook has determined that LAR 03-02 meets the criterion of 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Review (see Section VI of Enclosure 1).

The Station Operation Review Committee and the Company Nuclear Review Board have reviewed LAR 03-02.

A001

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Regulatory Programs Manager, at (603) 773-7194.

Very truly yours,
FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

cc:

H. J. Miller, NRC Regional Administrator
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Enclosure 1 to NYN-03061

FPL Energy
Seabrook Station

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

License Amendment Request 03-02
"Implementation of Alternate Source Term"

This License Amendment Request is submitted by FPL Energy Seabrook, LLC pursuant to the requirements of 10 CFR 50.4, 10 CFR 50.67 and 10CFR 50.90. The following information is enclosed in support of this License Amendment Request:

- **Section I - Introduction and Safety Assessment for Proposed Changes**
- **Section II - Markup of Proposed Changes**
- **Section III - Retype of Proposed Changes**
- **Section IV - Determination of Significant Hazards for Proposed Changes**
- **Section V - Proposed Schedule for License Amendment Issuance And Effectiveness**
- **Section VI - Environmental Impact Assessment**

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC hereby affirms that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Mark E. Warner
Site Vice President

Sworn and Subscribed
before me this
6th **day of** October **, 2003**

Notary Public

MICHAEL D. O'KEEFE
MY COMMISSION EXPIRES FEB. 5, 2008
NOTARY PUBLIC
NEW HAMPSHIRE

SECTION I

INTRODUCTION AND SAFETY ASSESSMENT FOR PROPOSED CHANGES

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

The current Seabrook Station licensing basis for the radiological consequences analyses for accidents discussed in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) is based on source term methodologies and assumptions derived from Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors (1962)."

Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10CFR 50.67 was issued by the NRC to permit holders of operating licenses to voluntarily revise the traditional accident source term used in the design basis accident radiological consequence analyses with Alternative Source Terms (ASTs). Part 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of the affected design basis accidents. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.

As documented in Revision 1 of NEI 99-03, "Control Room Habitability Guidance" several nuclear plants performed testing on control room unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in the accident analyses. NRC Generic Letter 2003-01, Control Room Habitability, discusses this issue further and requires that utilities assess the most limiting unfiltered inleakage into the Control Room Envelope (CRE). The AST methodology as established in RG 1.183 (July 2000) is being used to calculate the offsite and control room radiological consequences for Seabrook Station to support a power uprate and the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage.

The following UFSAR Chapter 15 accidents are analyzed:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor)
- Rod Cluster Control Assembly (RCCA) Ejection
- Failure of Small Lines Carrying Primary Coolant Outside Containment (Letdown Line Break)
- Radioactive Gaseous Waste System Leak or Failure, and
- Radioactive Liquid Waste System Leak or Failure (release to atmosphere).

Each accident and the specific input assumptions are described in "AST Licensing Technical Report for Seabrook Station" provided in Enclosure 2. These analyses provide for a bounding allowable control room unfiltered air inleakage of 150 cfm.

B. Description of Proposed Amendment

FPLE Seabrook proposes to revise the Seabrook Station licensing basis to implement AST, described in RG 1.183 (July 2000), through reanalysis of the radiological consequences of the UFSAR Chapter 15 accidents listed above. As part of the full implementation of this AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10CFR50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11.
- New onsite (Control Room) and offsite atmospheric dispersion factors are developed.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.
- Increased values for control room unfiltered air inleakage are assumed (unfiltered inleakage increased until applicable dose limit is approached).

Accordingly, the following change to the Seabrook Station, Unit No. 1, Technical Specifications (TS) is proposed:

- The definition of Dose Equivalent I-131 in Section 1.10 is revised to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of thyroid dose conversion factors.

Accident Source Term

The full core isotopic inventory for Seabrook Station is determined in accordance with RG 1.183 (July 2000). A planned power uprate is considered for conservatism. The inventory of fission products in the core and coolant systems that is available for release to the containment is based on the maximum expected power operation of the core and the expected values for fuel enrichment and fuel burnup. Event-specific isotopic source terms are developed using a bounding approach. The maximum uprated core power of 3659 MW_{th} is used, which considers calorimetric uncertainty. The period of irradiation is selected to be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.

The core inventory release fractions for the gap release and early in-vessel damage phases for the design basis LOCA utilized those release fractions provided in RG 1.183 (July 2000), Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released into Containment." For non-LOCA events, the fractions of the core inventory assumed to be in the gap are consistent with RG 1.183 (July 2000), Regulatory Position 3.2, Table 3,"Non-LOCA Fraction of Fission Product

Inventory in Gap." In some cases, the gap fractions listed in Table 3 are modified as required by the event-specific source term requirements listed in the Appendices for RG 1.183 (July 2000).

The nominal primary coolant activity is based on 1% failed fuel. The iodine activities are adjusted to achieve the Technical Specification limit of 1.0 μCi/gm dose equivalent I-131 using the proposed Technical Specification definition of Dose Equivalent I-131 (DE I-131) and dose conversion factors for individual isotopes from FGR 11. The remaining (non-iodine) isotopes are adjusted to achieve the Technical Specification limit of 100/E-bar microcuries per gram of gross activity.

Secondary coolant system activity is limited to a value of ≤ 0.10 μCi/gm dose equivalent I-131 in accordance with the Technical Specifications. Noble gases entering the secondary coolant system from the primary coolant system are assumed to be immediately released; thus the noble gas activity concentration in the secondary coolant system is assumed to be 0.0 μCi/gm. Therefore, the secondary side iodine activity is 1/10 of the primary coolant activity.

The fuel handling accident for Seabrook Station assumes the failure of one assembly; therefore, the fuel handling accident source term is based on a single "bounding" fuel assembly. Sensitivity studies were performed to assess the bounding fuel enrichment and bounding burnup values. The assembly source term is based on the planned uprated power plus calorimetric uncertainty (3659 MW_{th}). For each nuclide, the bounding activity for the allowable range of enrichments and discharge exposure is determined.

The AST Licensing Technical Report for Seabrook Station provides the details of the LOCA and non-LOCA accident analyses performed according to the guidelines set forth in RG 1.183 (July 2000).

Dose Calculation

The Seabrook Station dose calculations using the AST methodology apply TEDE acceptance criteria. Dose calculations follow the guidelines of Regulatory Positions cited in RG 1.183 (July 2000).

Analyses consider the radionuclides listed in Table 5 of RG 1.183 (July 2000) and assume that fission products are released to containment in particulate form, except for elemental iodine, organic iodine, and noble gases. Radioiodine fractions released to containment in a postulated accident are assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide, including both gap releases and fuel pellet releases. In specific instances, transport models may affect radioiodine fractions.

Assumptions and Methodologies

The AST analyses performed for Seabrook Station use assumptions and models defined in RG 1.183 (July 2000) to provide appropriate and prudent safety margins.

Except as otherwise stated, credit is taken for Engineered Safety Features (ESF) and other appropriately qualified, safety-related, accident mitigation features. Selected numeric input values are conservative to ensure a conservative calculated dose. Except as otherwise required by regulatory guidance, analyses use current licensing basis values and where more limiting, planned power uprate values.

Meteorological data collected per the Seabrook Station meteorological monitoring program described in the UFSAR is used in generating the accident atmospheric dispersion (*X/Q*) factors.

Dose Consequences Results

Full implementation of the Alternative Source Term methodology, as defined in Regulatory Guide 1.183 (July 2000), into the design basis accident analysis is made to support a planned power uprate and control room habitability in the event of increases in control room unfiltered air inleakage. Analysis of the dose consequences of the Loss-of-Coolant Accident (LOCA), Fuel Handling Accident (FHA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Reactor Coolant Pump Shaft Seizure (Locked Rotor), Rod Cluster Control Assembly (RCCA) Ejection, Small Line Break Outside Containment (Letdown Line Break), Radioactive Gaseous Waste System Failure and Radioactive Liquid Waste System Failure are made using the RG 1.183 (July 2000) methodology. Where RG 1.183 (July 2000) guidance does not exist, existing licensing basis and Standard Review Plan guidance are used.

C. Safety Assessment of Proposed Changes

The safety assessment of the proposed changes to the Technical Specification and the Seabrook Station licensing bases are described in Numerical Applications Inc., Report Number NAI-1131-013, Revision 2, "AST Licensing Technical Report for Seabrook Station," provided in Enclosure 2.

<u>**Conclusion:**</u>

In conclusion, based on the considerations discussed above and in the Alternate Source Term Technical Report for Seabrook Station, 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the Commission's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

SECTION II

MARKUP OF PROPOSED CHANGES

Refer to the attached markup of the proposed changes to the Technical Specifications. The attached markup reflects the currently issued revision of the Technical Specifications listed below. Pending Technical Specifications or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

The following Technical Specification change is included in the attached markup:

Technical Specification	Title	Page
1.12	Dose Equivalent I-131	1-3

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I"~~ Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

1.14 The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlappping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

SECTION III

RETYPE OF PROPOSED CHANGES

Refer to the attached retype of the proposed changes to the Technical Specifications. The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

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a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

SECTION IV

DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

FPLE Seabrook proposes to revise the Seabrook Station licensing basis to implement an Alternate Source Term, described in Regulatory Guide 1.183, (July 2000) through reanalysis of the radiological consequences of the following UFSAR Chapter 15 accidents:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor)
- Rod Cluster Control Assembly (RCCA) Ejection
- Failure of Small Lines Carrying Primary Coolant Outside Containment (Letdown Line Break)
- Radioactive Gaseous Waste System Leak or Failure, and
- Radioactive Liquid Waste System Leak or Failure (release to atmosphere).

As part of the full implementation of this AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10CFR50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11.
- New onsite (Control Room) and offsite atmospheric dispersion factors are developed.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.
- Increased values for control room unfiltered air inleakage are assumed (unfiltered inleakage increased until applicable dose limit is approached). The AST methodology as established in RG 1.183 (July 2000) is being used to calculate the offsite and control room radiological consequences for Seabrook Station to support the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage.

The full implementation of AST is supported by the following Technical Specification change:

The definition of Dose Equivalent I-131 in Section 1.12 is revised to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of thyroid dose conversion factors.

In accordance with 10 CFR 50.92, FPLE Seabrook has concluded that the proposed changes do not involve a significant hazards consideration (SHC). The basis for the conclusion that the proposed changes do not involve a SHC is as follows:

1. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   Alternative source term calculations have been performed that demonstrate the dose consequences remain below limits specified in NRC Regulatory Guide 1.183 (July 2000) and 10CFR50.67. The proposed change does not modify the physical design or operation of the plant. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of the accident. AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed change to the Technical Specifications, demonstrate that the dose consequences of these limiting events are all within the regulatory limits. The proposed Technical Specification change to the definition of dose equivalent I-131 is consistent with the implementation of AST and the requirements of RG 1.183 (July 2000).

   Therefore, the proposed change does not involve a significant increase the probability or consequences of an accident previously evaluated.

2. *The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.*

   The proposed change does not affect any plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change. The alternative source term and the dose equivalent I-131 definition change do not have the capability to initiate accidents. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed changes do not involve a significant reduction in the margin of safety.*

The proposed implementation of the alternative source term methodology is consistent with NRC RG 1.183 (July 2000). The Technical Specification change to the definition of dose equivalent I-131 is consistent with the implementation of AST and the requirements of RG 1.183 (July 2000). Conservative methodologies, per the guidance of RG 1.183 (July 2000), have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with use of the alternative source term methodology.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries and in the Control Room are within the corresponding regulatory limits of RG 1.183 (July 2000) and 10CFR50.67. The margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are set at or below the 10CFR50.67 limits. An acceptable margin of safety is inherent in these limits.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above evaluation, FPLE Seabrook concludes that the proposed change does not constitute a significant hazard.

SECTIONS V AND VI

PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

AND

ENVIRONMENTAL IMPACT ASSESSMENT

V. PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

FPLE Seabrook requests NRC review of License Amendment Request 03-02, and issuance of a license amendment by September 25, 2004, having immediate effectiveness and implementation within 60 days.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

FPLE Seabrook has reviewed the proposed license amendment against the criteria of 10 CFR 51.22 for environmental considerations. 10CFR51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

FPLE Seabrook has reviewed this proposed license amendment request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment. FPLE Seabrook proposes to revise the UFSAR Chapter 15 accident analyses to adopt the alternative source term methodology using the guidance of NRC Regulatory Guide 1.183 (July 2000).

This change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9) for the following reasons:

1. As demonstrated in the Section IV of this LAR, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite.

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change is analytical and does not result in any physical plant changes or new surveillance that would significantly increase the

cumulative occupational radiation exposure. Therefore, the proposed amendment has no significant affect on either individual or cumulative occupational radiation exposure.

Enclosure 2 to NYN-03061

NAI Report Release

Report Number: NAI-1131-013

Revision Number: 2

Title: AST Licensing Technical Report for Seabrook Station

Description:

> This report documents the results of the analyses and evaluations performed by Numerical Applications, Inc. in support of the Seabrook project to implement alternative radiological source terms. Design basis accidents and radiological consequences are evaluated using the AST methodology to support control room habitability in the event of increases in unfiltered inleakage. The analyses and evaluations performed by NAI are based on the guidance of Regulatory Guide 1.183.

Author (Jim Harrell)	9/8/03 Date
Reviewer (Steve Thomasson)	9/8/03 Date
Reviewer (Merv Marshall)	9/8/03 Date
Reviewer (Joe Sinodis)	9/8/03 Date
NAI Management (Tom George)	Sept 8, 2003 Date

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Figures and Tables

1.0 Radiological Consequences Utilizing the Alternative Source Term Methodology

1.1 Introduction

The current Seabrook Station licensing basis for the radiological consequences analyses for accidents discussed in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) is based on methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 and other early guidance.

Regulatory Guide (RG) 1.183 provides guidance on application of Alternative Source Terms (AST) in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10CFR50.67. Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10CFR50.67 allows holders of operating licenses to voluntarily revise the traditional accident source term used in the design basis accident (DBA) radiological consequence analyses with alternative source terms (ASTs).

The AST radiological consequence analyses in this report were performed with inputs intended to bound future implementation of power uprate by Seabrook Station up to 3659 MW_{th} (this value includes calorimetric uncertainties).

1.2 Evaluation Overview and Objective

As documented in NEI 99-03 and Generic Letter 2003-01, several nuclear plants performed testing on control room unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in their current accident analyses. The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for Seabrook Station to support the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage. This report represents a full implementation of the Alternative Source Term.

The following limiting UFSAR Chapter 15 accidents are analyzed:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor)
- Rod Cluster Control Assembly (RCCA) Ejection
- Failure of Small Lines Carrying Primary Coolant Outside Containment (Letdown Line Break)
- Radioactive Gaseous Waste System Leak or Failure, and
- Radioactive Liquid Waste System Leak or Failure (release to atmosphere).

Each accident and the specific input and assumptions are described in Section 2.0 of this report. These analyses provide for a bounding allowable control room unfiltered air inleakage of 150 cfm. The use of 150 cfm as a design basis value is expected to be above the unfiltered inleakage value to be determined through testing and analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01.

1.3 Proposed Changes to the Seabrook Station Licensing Basis

FPL Energy proposes to revise the Seabrook Station licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences of the UFSAR Chapter 15 accidents listed in Section 1.2 above. As part of the full implementation of this AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10CFR50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11.
- New onsite (Control Room) and offsite atmospheric dispersion factors are developed.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.
- Increased values for control room unfiltered air inleakage are assumed (unfiltered inleakage increased until applicable dose limit is approached).

Accordingly, the following change to the Seabrook Station Technical Specifications (TS) is proposed:

- The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of thyroid dose conversion factors. Use of thyroid dose conversion factors (versus effective dose conversion factors for inhalation or CEDE doses) resulted in slightly more conservative total iodine concentrations in the primary coolant and, therefore, slightly higher doses. Precedent for using thyroid dose conversion factors from FGR 11 is established in the Shearon Harris Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 107 to NPF-63 issued October 12, 2001 (specifically page 22 of the Safety Evaluation Report).

1.4 Compliance with Regulatory Guidelines

The revised Seabrook Station accident analyses addressed in this report follow the guidance provided in RG 1.183. Assumptions and methods utilized in this analysis for which no specific guidance is provided in RG 1.183, but for which a regulatory precedent has been established, are as follows:

- Selection of the Radioactive Gaseous and Liquid System Failures and Letdown Line Break dose consequences acceptance criteria for the EAB and LPZ are based on the current licensing basis of "a small fraction" of the guidelines, which is defined as 10%. 10% of the 10CFR50.67 limits for the EAB and LPZ equals 2.5 rem TEDE.

- Use of the MicroShield code to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis. It is qualified for this application and has been used to support licensing submittals that have been accepted by the NRC. Precedent for this use of MicroShield is established in the Duane Arnold Energy Center submittal dated October 19, 2000 and associated NRC Safety Evaluation dated July 31, 2001.

Guidance from other regulatory guides, such as RG 1.52, RG 1.145 and RG 1.194, is also used in the development of the revised analyses.

1.5 Computer Codes

The following computer codes are used in performing the Alternative Source Term analyses:

Computer Code	Version	Reference	Purpose
ARCON96	June 1997	5.14	Atmospheric Dispersion Factors
MicroShield	5.05	5.15	Direct Shine Dose Calculations
ORIGEN	2.1	5.16	Core Fission Product Inventory
PAVAN	2.0	5.17	Atmospheric Dispersion Factors
RADTRAD-NAI	1.1	5.18	Radiological Dose Calculations

1.5.1 ARCON96 – used to calculate relative concentrations (*X/Q* factors) in plumes from nuclear power plants at control room intakes in the vicinity of the release point using plant meteorological data. Qualified for this application under NAI's 10CFR50 Appendix B program.

1.5.2 MicroShield – used to analyze shielding and estimate exposure from gamma radiation. Qualified for this application under NAI's 10CFR50 Appendix B program.

1.5.3 ORIGEN – used for calculating the buildup, decay, and processing of radioactive materials. Qualified for this application under NAI's 10CFR50 Appendix B program.

1.5.4 PAVAN – provides relative air concentration (*X/Q*) values as functions of direction for various time periods at the EAB and LPZ boundaries assuming ground-level releases or elevated releases from freestanding stacks. Qualified for this application under NAI's 10CFR50 Appendix B program.

1.5.5 RADTRAD-NAI – estimates the radiological doses at offsite locations and in the control room of nuclear power plants as consequences of postulated accidents. The code considers the timing, physical form (i.e., vapor or aerosol) and chemical species of the radioactive material released into the environment.

RADTRAD-NAI began with versions 3.01 and 3.02 of the NRC's RADTRAD computer code, originally developed by Sandia National Laboratory (SNL). The code is initially modified to compile on a UNIX system. Once compiled, an extensive design review/verification and validation process began on the code and documentation. The subject of the review also included the source code for the solver, which is made available in a separate distribution from the NRC. RADTRAD-NAI validation is performed with three different types of tests:

- Comparison of selected Acceptance Test Case results with Excel spreadsheet solutions and hand solutions,
- Separate effects tests, and
- Industry examples.
- The industry examples included prior AST submittals by BWRs and PWRs, as well as other plant examples.

In addition to reviewing the code and incorporating error corrections, several software revisions were made. One revision involved the consideration of noble gases generated by decay of isotopes on filters that are returned to the downstream compartment. Another revision involved the modification of the dose conversion and nuclide inventory files to account for 107 isotopes to assure that significant dose contributors were addressed. The dose conversion factors used by RADTRAD-NAI are from Federal Guidance Report Nos. 11 and 12 (FGR 11and FGR 12).

Multiple control room atmospheric dispersion factors for different release-receptor combinations are allowed in Version 1.1, which also prints nuclide inventories on filters.

RADTRAD-NAI was developed and is maintained under Numerical Applications' 10CFR50 Appendix B program.

1.6 Radiological Evaluation Methodology

1.6.1 Analysis Input Assumptions

Common analysis input assumptions include those for the control room ventilation system and dose calculation model (Section 1.6.3), direct shine dose (Section 1.6.5), radiation source terms (Section 1.7), and atmospheric dispersion factors (Section 1.8). Event-specific assumptions are discussed in the event analyses in Section 2.0.

1.6.2 Acceptance Criteria

Offsite and Control Room doses must meet the guidelines of RG 1.183 and requirements of 10CFR50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of RG 1.183. For analyzed events not addressed in RG 1.183, the basis used to establish the acceptance criteria for the radiological consequences is provided in the discussion of the event in Section 2.0. For Seabrook Station, the events not specifically addressed in RG 1.183 are the Radioactive Gaseous and Liquid System Ruptures and Letdown Line Break.

1.6.3 Control Room Ventilation System Description

The Normal Makeup Air Subsystem consists of two 100 percent capacity vane axial fans with a flow capacity of 1000 cfm each at the system static pressure, and the associated dampers. Air is drawn from two remote air intakes (east and west located more than 700 ft. apart). Location of the air intakes was selected considering the plant configuration and the site-specific meteorological conditions to preclude contamination of both intakes at the same time. Air flows through two 12" heavy wall carbon steel pipes provided with radiation and smoke detecting devices, as well as a normally open manual isolation valve on each path. Two 18" lines, each provided with a backdraft damper bypass the normal makeup air supply fans to supply makeup air to the filtration assemblies during the emergency mode of operation.

The Emergency Makeup Air and Filtration Subsystem consists of two filtration assemblies with a maximum capacity of 1210 cfm each (= 1100 cfm + 10% tolerance). Each assembly includes a prefilter, an electric heater, a HEPA-Carbon-HEPA filter configuration, a fan, manual inlet isolation damper, discharge isolation dampers and backdraft dampers.

In the event of an accident with a significant radiological release, high radiation is detected in either remote air supply piping, the Emergency Makeup Air and Filtration Subsystem fans are actuated and their associated dampers (1-CBA-DP-27A and DP-27B) are opened. The normal makeup air fan automatically trips off and its associated discharge damper, is automatically closed. The isolation function of these dampers is safety-related.

When the normal makeup air flow path is isolated, air is drawn from the remote air intakes through the bypass lines provided with backdraft dampers. When the Emergency Makeup Air and Filtration Subsystem fans are actuated, they generate a Control Room Makeup Air Filter Recirculation Mode Signal.

Although the redundant filtration assembly fans are capable of operating simultaneously, plant operators may decide to shut down one of the fans during the course of the accident.

Each filtration assembly has an average flow capacity of 1100 cfm consisting of 600 cfm of outside air (half of it from each intake) and the remainder is recirculating air. With two-fan operation, the average flow capacity increases to 1370 cfm consisting of 970 cfm of outside air with the rest being recirculated air.

The operating filtration assembly draws makeup air into the suction plenum, through the prefilter and the heater and then mixes with recirculation air drawn from the Mechanical Equipment Room. The mixed air flows through the HEPA-carbon-HEPA filters before it is discharged into the Mechanical Equipment Room by the filter fan.

1.6.3.1 Control Room Dose Calculation Model

The Control Room model includes a recirculation filter model along with filtered air intake, unfiltered air inleakage (typically through one or two inleakage paths that bound the expected inleakage locations) and an exhaust path. System performance, sequence, and timing of operational evolutions associated with the CR ventilation system are discussed below. Control Room ventilation system parameters assumed in the analyses are provided in Table 1.6.3-1. The dispersion factors for use in modeling the Control Room during each mode of operation are provided in Tables 1.8.1-2 and 1.8.1-3. Control Room occupancy factors and assumed breathing rates are those prescribed in RG 1.183. Figure 1.8.1-1 provides a site sketch showing the Seabrook Station layout, including the location of onsite potential radiological release points with respect to the control room air intakes. The elevations of release points and intakes used in the Control Room AST dose assessments are provided in Table 1.8.1-1.

The control room ventilation system contains a filtration system for removal of radioactive iodine and particulate material that may enter the CR during the course of the event. Calculation of the dose to operators in the control room requires modeling of various system configurations and operating evolutions of the control room ventilation system during the course of the accident. The control room model will define at least two concurrent air intake paths representing the defined CR ventilation system air intake and the unfiltered inleakage into the CR. Outside air can enter the control room through the filtration/ventilation system from both of two ventilation intake locations. Due to their diverse locations, these intakes are assigned different dispersion factors for calculating the concentration of radioactive isotopes in the air drawn in through that intake due to the activity released from various locations on the site during an accident. Unfiltered outside air can also enter the CR directly from various sources of unfiltered inleakage. Modeling of the Control Room will address these factors as they apply to the various release locations for each analyzed event. Details of the CR modeling for each event are described in subsequent event analyses sections.

Three main pathways for unfiltered inleakage to the control room were considered; inleakage via the diesel building, inleakage via the primary control room entrance (double air lock configuration), and inleakage via the emergency fire exit (two doors in series). A value of 10 cfm is typically assumed for door leakage for normal ingress/egress. However, this flow would be reduced or eliminated by a two-door vestibule. It was conservatively assumed that 20 cfm of total door leakage occurs via the most limiting door. The X/Qs for the fire exit are always more limiting than those for the primary control room entrance; therefore, all of the unfiltered inleakage via the doors was assumed to occur at the fire exit. For most release locations, the X/Qs for the fire exit are more limiting than the X/Qs for the diesel building inleakage. For these cases, the fire exit and the diesel building were considered as separate paths for unfiltered inleakage. In cases where the diesel building is more limiting than the fire exit, all of the unfiltered inleakage was assumed to enter via the diesel building.

With the control room inleakage rates considered in the Alternative Source Term analyses in this report, single fan operation is limiting with respect to radiological consequences.

1.6.4 Control Room Inleakage Sensitivity Study

The results of the control room dose calculations were used to establish the sensitivity of the control room dose due the amount of "unfiltered inleakage" assumed to be introduced into the control room. Sensitivity studies were performed that varied allowances for unfiltered control room air inleakage. The results were then used to establish the maximum allowable unfiltered CR inleakage.

The event-specific modeling assumptions used to construct the RADTRAD-NAI files for performing the various aspects of the accident dose calculation are discussed in subsequent event analysis sections along with the input parameters used to model the Seabrook Station plant parameters. The cases presented represent the cases using the control room unfiltered inleakage rate that is determined by the sensitivity study to be limiting with respect to the CR dose acceptance criteria. The limiting unfiltered CR inleakage rates assumed in the analyses are provided in Table 1.6.3-1.

1.6.5 Direct Shine Dose

The total control room dose also requires the calculation of direct shine dose contributions from:

- the radioactive material on the control room filters ,
- the radioactive plume in the environment, and
- the activity in the primary containment atmosphere.

The contribution to the total dose to the operators from direct radiation sources such as the control room filters, the containment atmosphere, and the released radioactive plume were calculated for the LOCA event. The LOCA shine dose contribution is assumed to be bounding for all other events. The 30-day direct shine dose to a person in the control room, considering occupancy, is provided in Table 1.6.3-2.

Direct shine dose is determined from three different sources to the control room operator after a postulated LOCA event. These sources are the containment, the control room air filters, and the external cloud that envelops the control room. All other sources of direct shine dose are considered negligible. The MicroShield 5 code is used to determine direct shine exposure to a dose point located in the control room. Each source required a different MicroShield case structure including different geometries, sources, and materials. The external cloud is assumed to have a length of 1000 meters in the MicroShield cases to approximate an infinite cloud. A series of cases is run with each structure to determine an exposure rate from the radiological source at given points in time. These sources were taken from RADTRAD-NAI runs that output the nuclide activity at a given point in time for the event. The RADTRAD-NAI output provides the time dependent results of the radioactivity retained in the control room filter components, as well as the activity inventory in the environment and the containment. A bounding CR filter inventory is established using a case from the sensitivity study with unfiltered inleakage that produced a control room dose slightly in excess of the 5 rem TEDE dose limit to control room operators. The direct shine dose calculated due to the filter loading for this conservative unfiltered inleakage case is used as a conservative assessment of the direct shine dose contribution for all accidents.

The RADTRAD-NAI sources were then input into the MicroShield case file. The exposure results from the series of cases for each source term were then corrected for occupancy using the occupancy factors specified in RG 1.183. The cumulative exposure and dose are subsequently calculated to yield the total 30-day direct shine dose from each source. The results of the Direct Shine Dose evaluation are presented in Table 1.6.3-2.

1.7 Radiation Source Terms

1.7.1 Fission Product Inventory

The source term data to be used in performing alternative source term (AST) analyses for Seabrook Station are summarized in the following tables:

Table 1.7.2-1 - Primary Coolant Source Term
Table 1.7.3-1 - Secondary Side Source Term (non-LOCA)
Table 1.7.4-1 - LOCA Containment Leakage Source Term
Table 1.7.5-1 - Fuel Handling Accident Source Term

Note that the source terms provided in the referenced tables do not include any decay before the start of the events. Decay time assumptions are applied in the RADTRAD-NAI cases for individual event analysis. For example, the RADTRAD-NAI case for the Fuel Handling Accident analysis would account for the required decay time before the movement of fuel is allowed (as determined by Technical Specifications).

The Seabrook Station reactor core consists of 193 fuel assemblies. The full core isotopic inventory is determined in accordance with RG 1.183, Regulatory Position 3.1, using the ORIGEN-2.1 isotope generation and depletion computer code to develop the isotopics for the specified burnup, enrichment, and burnup rates (power levels). The plant-specific isotopic source terms are developed using a bounding approach.

Sensitivity studies were performed to assess the bounding fuel enrichment and bounding burnup values. The assembly source term is based on uprated power with calorimetric uncertainty (3659 MW_{th}). For rod average burnups in excess of 54,000 MWD/MTU the heat generation rate is limited to 6.3 kw/ft in accordance with RG 1.183. For non-LOCA events with fuel failures, a bounding radial peaking factor of 1.65 is then applied to conservatively simulate the effect of power level differences across the core that might affect the localized fuel failures for assemblies containing the peak fission product inventory.

The core inventory release fractions for the gap release and early in-vessel damage phases for the design basis LOCAs utilized those release fractions provided in RG 1.183, Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released into Containment." For non-LOCA events, the fractions of the core inventory assumed to be in the gap are consistent with RG 1.183, Regulatory Position 3.2, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap." In some cases, the gap fractions listed in Table 3 are modified as required by the event-specific source term requirements listed in the Appendices for RG 1.183.

The following assumptions are applied to the source term calculations:

1. A conservative maximum fuel assembly uranium loading (492 kilograms) is assumed to apply to all 193 fuel assemblies in the core.

2. Radioactive decay of fission products during refueling outages is ignored in the source term calculation.

3. When adjusting the primary coolant isotopic concentrations to achieve Technical Specification limits, the relative concentrations of fission products in the primary coolant system are assumed to remain constant.

1.7.2 Primary Coolant Source Term

The primary coolant source term for Seabrook Station is derived from Table 11.1-1 of the UFSAR. The activities from the column for 1% clad defects from UFSAR Table 11.1-1 are used. UFSAR Table 11.1-2 summarizes the parameters used in the calculation of the primary coolant source term.

The iodine activities from UFSAR Table 11.1-1 are adjusted to achieve the Technical Specification 3.4.8 limit of 1.0 μCi/gm dose equivalent I-131 using the proposed Technical Specification definition of Dose Equivalent I-131 (DE I-131) and dose conversion factors for individual isotopes from FGR 11. The non-iodine species are adjusted to achieve the Technical Specification limit of 100/E-bar for non-iodine activities.

The dose conversion factors for inhalation and submersion are from Federal Guidance Reports Nos. 11 and 12 respectively.

The final adjusted primary coolant source term is presented in Table 1.7.2-1,"Primary Coolant Source Term."

1.7.3 Secondary Side Coolant Source Term

Secondary coolant system activity is limited to a value of ≤ 0.10 μCi/gm dose equivalent I-131 in accordance with TS 3.7.1.4. Noble gases entering the secondary coolant system are assumed to be immediately released; thus the noble gas activity concentration in the secondary coolant system is assumed to be 0.0 μCi/gm. Thus, the secondary side iodine activity is 10% of the activity given in Table 1.7.2-1.

The secondary side source term is presented in Table 1.7.3-1, "Secondary Side Source Term."

1.7.4 LOCA Containment Leakage Source Term

Per Section 3.1 of Reg. Guide 1.183, the inventory of fission products in the Seabrook Station reactor core and available for release to the containment is based on the maximum expected uprated power operation of the core plus calorimetric uncertainty (3659 MW_{th}) and the associated limiting values for fuel enrichment and fuel burnup. The period of irradiation is selected to be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. In addition, for the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory is used.

During a LOCA, all of the fuel assemblies are assumed to fail; therefore, the source term is based on an "average" assembly with a core average burnup of 45,000 MWD/MTU and an average assembly power* of 18.96 MW_{th}. The minimum fuel enrichment is based on an historical minimum of 1.6 w/o and the maximum fuel enrichment is the Technical Specification maximum value of 5.0 w/o. It is conservatively assumed that a maximum assembly uranium mass of 492,000 gm applies to all of the fuel assemblies.

*Average assembly power = (3659 MW_{th})(1 / 193 assemblies) = 18.96 MW_{th} / assembly

The ORIGEN runs used cross section libraries that correspond to PWR extended burnup fuel. Decay time between cycles is conservatively ignored. For each nuclide, the bounding activity for the allowable range of enrichments is determined.

The LOCA source term is presented in Table 1.7.4-1,"LOCA Containment Leakage Source Term."

1.7.5 Fuel Handling Accident Source Term

The fuel handling accident for Seabrook Station assumes the failure of one assembly; therefore, the fuel handling accident source term is based on a single "bounding" fuel assembly.

Per Section 3.1 of Reg. Guide 1.183, the source term methodology for the Fuel Handling Accident is similar to that used for developing the LOCA containment leakage source term, except that for DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, a radial peaking factor of 1.65 is applied in determining the inventory of the damaged rods.

The LOCA containment leakage source term is based on the activity of 193 fuel assemblies. The radial peaking factor is 1.65. Thus, based on the methodology specified in Reg. Guide 1.183, the fuel handling accident source term is derived by applying a factor of 1.65/193 to the LOCA containment leakage source term. To ensure that the "bounding" assembly is identified, the activity of a peak burnup assembly (62,000 MWD/MTU), at 1.6 w/o, 3.8 w/o and 5.0 w/o, is determined and compared to the source term derived from the LOCA data. For each nuclide, the bounding activity for the allowable range of enrichments and discharge exposure is determined.

The FHA source term is presented in Table 1.7.5-1,"Fuel Handling Accident Source Term."

1.8 <u>Atmospheric Dispersion (*X/Q*) Factors</u>

1.8.1 Onsite *X/Q* Determination

New *X/Q* factors for onsite release-receptor combinations are developed using the ARCON96 computer code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664). Additionally, NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003, has been implemented. Regulatory Guide 1.194 contains new guidance that supercedes the NUREG/CR-6331 recommendations for using certain default parameters as input. Therefore, the following changes from the default values are made:

- For surface roughness length, m, a value of 0.2 is used in lieu of the default value of 0.1, and
- For averaging sector width constant, a value of 4.3 is used in lieu of the default value of 4.0.

A number of various release-receptor combinations are considered for the onsite control room atmospheric dispersion factors. These different cases are considered to determine the limiting release-receptor combination for the events. The limiting release-receptor location combinations are the following:

- Plant Vent to Control Room receptor point,
- Closest Containment Surface to Control Room receptor point,
- RWST to Control Room receptor point,
- Containment Personnel Hatch to Control Room receptor point,
- Main Steam Line Closest Point to Control Room receptor point,
- Main Steam Line Chase Panel to Control Room receptor point,
- Closest MSSV to Control Room receptor point,
- Closest ARV to Control Room receptor point,
- Primary Auxiliary Building Louver to Control Room receptor point,
- Primary Auxiliary Building Fan to Control Room receptor point,

- Turbine Building Closest Point to Control Room receptor point,
- Waste Process Building SW Corner Roll-Up Door to Control Room receptor point,
- Carbon Delay Bed to Control Room receptor point, and
- BWST to Control Room receptor point.

Figure 1.8.1-1 provides a sketch of the general layout of Seabrook that has been annotated to highlight the release and receptor point locations described above, among others. All releases are taken as ground releases per guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants," Rev. 1, February 1983.

Table 1.8.1-1, "Release-Receptor Combination Parameters for Analysis Events," provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release point. Angles are calculated based on trigonometric layout of release and receptor points in relation to the North-South and East-West axes. Direction values are corrected for "plant North" offset from "true North" by 35° 34' 0".

Table 1.8.1-2, "Onsite Atmospheric Dispersion Factors (*X/Q*) for Analysis Events," provides the Control Room *X/Q* factors for the release-receptor combinations listed above. These factors are not corrected for occupancy.

This table summarizes the *X/Q* factors for the control room intakes used in the various accident scenarios for onsite control room dose consequence analyses. Values are presented for the control room intake and the possible inleakage points through the Diesel Building or through the control room fire exit door. The fire exit door refers to the combination of the stairwell and kitchen door entrances on the south side of the control building. The *X/Q* values include taking credit for dilution because of the presence of dual intakes where allowed by Regulatory Guide 1.194. Based on the layout of the site, the only cases that may take credit for dilution are when the releases are to the normal and emergency control room intakes. The release from the RWST to the Diesel Building intakes utilize Equation 5a of Regulatory Guide 1.194 to treat the intakes on the north and south sides of the building as dual intakes.

Table 1.8.1-3, "Release-Receptor Point Pairs Assumed for Analysis Events," identifies the Release-Receptor pair and associated Control Room *X/Q* factors from Table 1.8.1-2, that are used in the event analyses for each of the three possible control room intake points.

1.8.2 Offsite *X/Q* Determination

For offsite receptor locations, the new atmospheric dispersion (*X/Q*) factors are developed using the PAVAN computer code ("PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations," NUREG/CR-2858, November 1982, RSICC Computer Code Collection No. CCC-445). The offsite maximum *X/Q* factors for the EAB and LPZ are presented in Table 1.8.2-1, "Offsite Atmospheric Dispersion Factors (*X/Q*)." In accordance with Regulatory Position 4 from NUREG/CR-2858, the maximum value from all downwind sectors for each time period is compared with the 5% overall site *X/Q* values for those boundaries, and the larger of the values is used in evaluations.

All of the releases are considered ground level releases because the highest possible release height (from the plant vent) is 185 feet. From Section 1.3.2 of RG 1.145, a release is only considered a stack release if the release point is at a level higher than two and one-half times the height of adjacent solid structures. For the Seabrook plant, the elevation of the top of the containment structure is 199.25 ft. The highest possible release point is not 2.5 times higher than the adjacent containment building; therefore, all releases are considered ground level releases. As such, the release height is set equal to 10.0 meters as required by Table 3.1 of NUREG/CR-2858. The building area used for the building wake term is the same as for some of the ARCON96 onsite *X/Q* cases. This area of 2,416 m^2 is

calculated to be conservatively small in that the height used in the area calculation is from the bottom of the containment dome to the grade elevation. The containment height used in the building wake term is the containment top elevation minus the bottom (grade) elevation of 19 ft. Release Point elevations are provided in Table 1.8.1-1,"Release-Receptor Combination Parameter for Analysis Events."

The tower height at which the wind speeds are measured is 10.05 meters. The maximum windspeed in each windspeed category is chosen to match the raw joint frequency distribution meteorological data.

1.8.3 Meteorological Data

Meteorological data over a five-year period (1998 through 2002) is used in the development of the new *X/Q* factors used in the analysis. The Seabrook Plant Meteorological Monitoring System, complies with RG 1.23, "Onsite Meteorological Programs," 1972. The Meteorological Monitoring System is described in the Seabrook Plant Design Basis Document DBD-MET.1, "Meteorological Monitoring System," and the Onsite Meteorological Measurement Program is described in Section 2.3.3 of the Seabrook UFSAR.

Five years worth of meteorological data is used which meets the guidance set forth in Section 3.1 of Regulatory Guide 1.194. The meteorological data for 1998 through 2002 was provided in electronic format in comma delimited text files. These files were appended to each other in chronological order to create one file to be accessed by ARCON96.

ARCON96 analyzes the meteorological data file used and lists the total number of hours of data processed and the number of hours of missing data in the case output. A meteorological data recovery rate may be determined from this information. Since all of the Seabrook cases use the same meteorological data file, all of the cases in this analysis have the same data recovery rate. The ARCON96 files present the number of hours of data processed as 43,824 and the number of missing data hours as 520. This yields a meteorological data recovery rate of 98.8%. No regulatory guidance is provided in Regulatory Guide 1.194 and NUREG/CR-6331 on the valid meteorological data recovery rate required for use in determining onsite *X/Q* values. However, Regulatory Position C.5 of RG 1.23 requires a 90% data recovery threshold for measuring and capturing meteorological data. Clearly, the 98.8% valid meteorological data rate for the cases in this analysis exceeds the 90% data recovery limit set forth by RG 1.23. With a data recovery rate of 98.8% and a total of five years worth of data, the contents of the meteorological data file are representative of the long-term meteorological trends at the Seabrook site.

The meteorological data were also provided in annual joint frequency distribution format for 1998 through 2002. The joint frequency distribution file requires the annual meteorological data to be sorted into several classifications. This is accomplished by using three classifications that include wind direction, wind speed, and atmospheric stability class. The format for the file conforms to the format provided in Table 1 of RG 1.23, with the exceptions of a category for the variable wind direction and that the wind directions are listed from NNE to N instead of N to NNW. These data are provided for each year in terms of the number of hours of that year that fell into each classification category. A spreadsheet was used to merge the five years worth of data into one joint frequency distribution file.

The total values for each stability class are then transposed so that the rows correspond to the wind speed bins and the columns correspond to the wind directions. The wind directions are then ordered properly so that the first column corresponds to the north wind direction and the last column corresponds to the NNW direction as required by the PAVAN code. The final ordered numbers are used in the input file for PAVAN.

The average daily temperature swing used for RWST releases is determined from the 2001 ASHRAE Fundamentals handbook. The value of 18.2 °F for the dry bulb temperature swing from Portsmouth, NH was used as it is the closest listed location to the Seabrook site.

The met data used for the ARCON96 runs is manipulated to determine the 95th percentile wind speed at the MSSV and ARV release heights. Any data determined to be invalid is excluded. The 95th percentile wind speed for the MSSV release height is 16.72 miles per hour. The 95th percentile wind speed for the ARV release height is 16.81 miles per hour.

The exit velocity for the uncapped, vertically oriented MSSV is greater than 5 times the above listed 95th percentile wind speed for the first 2.5 hours of the various events. Therefore, a factor of 5 reduction was applied to the χ/Q values for the MSSV releases during this time period in the event analyses. The factor of 5 reduction is discussed in Section 6 of Regulatory Guide 1.194. The NRC staff has expressed its willingness to consider plume rise on page 7 of Attachment 1 to the letter dated January 15, 2002 from the Donald C. Cook plant to the NRC with the subject, "Donald C. Cook Nuclear Plants Units 1 and 2, Partial Response to Second Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability." Precedent for the use of the factor of 5 reduction on the χ/Q values due to plume rise is established in Issuance of Amendment No. 271 to DPR-58 and the Issuance of Amendment No. 252 to DPR-74 and the associated Safety Evaluation issued November 14, 2002. Page 17 of the D. C. Cook Safety Evaluation discusses that the NRC staff views the χ/Q values determined with the factor of 5 reduction due to plume rise as acceptable.

2.0 Radiological Consequences – Event Analyses

2.1 Loss of Coolant Accident (LOCA)

2.1.1 Background

This event is assumed to be caused by an abrupt failure of the main reactor coolant pipe and the ECCS fails to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity is released from the containment and from there, released to the environment by means of containment leakage and leakage from the ECCS. This event is described in the Section 15.6.5 of the UFSAR.

2.1.2 Compliance with RG 1.183 Regulatory Positions

The LOCA dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided in Table 1.7.4-1. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of RG 1.183.

2. Regulatory Position 2 - The sump pH is controlled at a value greater than 7.0 per UFSAR Section 6.5.2.2. Therefore, the chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.

3. Regulatory Position 3.1 - The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

4. Regulatory Position 3.2 - Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.23 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed (based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983) for all aerosols in the unsprayed regions with no credit of natural deposition of aerosols in the sprayed regions. Precedent for conservatism of the 0.1 hr^{-1} natural deposition removal coefficient was established in the Indian Point Nuclear Generating Unit No. 2 Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 211 to DPR-26 issued July 27, 2000. No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 – Containment spray provides coverage to 85.4% of the containment. Therefore, the Seabrook containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into sprayed and unsprayed regions. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

   The SRP limits the spray removal coefficient for elemental iodine to 20 hr^{-1}; therefore although a

higher value was calculated, 20 hr^{-1} was used for the elemental iodine spray removal coefficient. In addition, the SRP and Reg. Guide 1.183 specify a maximum decontamination factor of 200 for spray removal of elemental iodine. The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is based on the maximum airborne elemental iodine concentration in the containment. The time for the containment sprays to reach an elemental iodine decontamination factor of 200 was determined by running a containment leakage case without environment leakage paths. Radioactive decay and natural deposition of iodine were conservatively left on as removal mechanisms contributing to the decontamination factor. Due to mixing between the sprayed and unsprayed regions of containment, the iodine activity in both containment regions was included in the determination of the time required to reach a decontamination factor of 200. The decontamination factor for elemental iodine reaches 200 at just over 2.92 hours.

The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 5.75 hr^{-1}, the time of a DF of 50 is computed with the same model used to determine the elemental iodine DF of 200. The time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.56 hours.

6. Regulatory Position 3.4 - Reduction in airborne radioactivity in the containment by filter recirculation systems is not assumed in this analysis.

7. Regulatory Position 3.5 - Not applicable to Seabrook.

8. Regulatory Position 3.6 - Not applicable to Seabrook.

9. Regulatory Position 3.7 - A containment leak rate of 0.15% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.075% per day of the containment air.

10. Regulatory Position 3.8 - Routine containment purge is considered in this analysis. The purge release evaluation assumes that 100% of the radionuclide inventory in reactor coolant system liquid (based on the Technical Specification RCS equilibrium activity) is released to the containment at the initiation of the LOCA. The purge system is isolated before the onset of the gap release phase.

11. Regulatory Position 4.1 - Leakage from containment collected by the secondary containment is processed by ESF filters prior to an assumed ground level release.

12. Regulatory Position 4.2 - Leakage into the secondary containment is assumed to be released directly to the environment as a ground level release prior to drawdown of the secondary containment at 4.5 minutes.

13. Regulatory Position 4.3 – The containment enclosure emergency air cleaning system is credited as being capable of maintaining a negative pressure with respect to the outside environment considering the effect of high windspeeds and LOCA heat effects on the annulus as described in UFSAR Sections 6.5.1.1 and 6.5.1.3.

14. Regulatory Position 4.4 - No credit is taken for dilution in the secondary containment volume.

15. Regulatory Position 4.5 - 60% of the primary containment leakage is assumed to bypass the secondary containment. This bypass leakage is released from containment without filtration.

16. Regulatory Position 4.6 - The containment enclosure emergency air cleaning system is credited as meeting the requirements of RG 1.52 and Generic Letter 99-02 per UFSAR Section 6.5.1.3 and UFSAR Table 6.5-1.

17. Regulatory Position 5.1 - Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

18. Regulatory Position 5.2 - Leakage from the ESF system is taken as two times 24 gallons per day for a total leakage rate of 48 gallons per day. The leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continues for the 30-day duration. Backleakage to the Refueling Water Storage Tank is also considered separately as two times the measured leakage value of 0.47975 gpm for a total leakage rate of 0.9595 gpm.

19. Regulatory Position 5.3 - With the exception of the non-particulate iodines, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

20. Regulatory Position 5.4 – A flashing fraction of 4.7% was determined based on the temperature of the containment sump liquid at the time recirculation begins. The iodine available for release at the time recirculation begins is based on expected sump pH history and temperature (see the Release Inputs in the Methodology section below). All of the non-particulate iodine available for release is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days. For ECCS leakage back to the RWST, the analysis demonstrates that the temperature of the leaked fluid will cool below 212°F prior to release from the tank.

21. Regulatory Position 5.5 - The iodine available for release at the time recirculation begins is based on expected sump pH history and temperature (see the Release Inputs in the Methodology section below). All of the non-particulate iodine available for release is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days. For the ECCS leakage back to the RWST, the sump and RWST pH history and temperature are used to evaluate the amount of iodine that enters the RWST air space.

22. Regulatory Position 5.6 - The temperature and pH history of the sump and RWST are considered in determining the radioiodine available for release and the chemical form. Credit is taken for hold-up and dilution of activity in the RWST as allowed by Regulatory Position 5.6. No credit for ESF filtration of the RWST leakage is taken. Filtration of non-RWST ECCS leakage is credited.

23. Regulatory Position 6 - Not applicable to Seabrook.

24. Regulatory Position 7 - Containment purge is not considered as a means of combustible gas or pressure control in this analysis; however, the effect of routine containment purge before isolation is considered.

2.1.3 Methodology

For this event, the Control Room ventilation system cycles through two modes of operation (the operational modes are summarized in Table 1.6.3-1). Inputs and assumptions fall into three main categories: Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

For the purposes of the LOCA analyses, a major LOCA is defined as a rupture of the RCS piping, including the double-ended rupture of the largest piping in the RCS, or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low-pressure trip setpoint is

reached. A safety injection system signal is actuated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. The following measures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Release Inputs

The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Table 1.7.4-1. The source term represents end of cycle conditions assuming enveloping initial fuel enrichment and an average core burnup of 45,000 MWD/MTU.

For the first 24 hours, the containment is assumed to leak at a rate of 0.15% of the containment air per day. Per RG 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.075% /day based on the post-LOCA primary containment pressure history.

The ESF leakage to the auxiliary building is assumed to be 48 gpd based upon two times the current value of 24 gpd. The temperature of the leakage is based on the sump temperature at and after the time recirculation begins (255 °F in the sump at the time recirculation begins). The leakage is assumed to start at 26 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that all of the non-particulate iodine available for release is released from the leaked liquid. Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis. Although the sump pH indicates the iodine in the sump solution is all in the nonvolatile iodide or iodate form, this analysis conservatively assumes that the chemical form of the iodine in the sump water at the time of recirculation (26 minutes) is 98.85% aerosol, 1% elemental and 0.15% organic.

The ECCS backleakage to the RWST is assumed to be 0.9595 gpm. The leakage is assumed to start at 26 minutes into the event when recirculation starts and continue throughout the 30-day period. Note that based on the leakage rate and the size of the piping, the leakage would not reach the RWST for an extended period of time after recirculation begins. This time period is conservatively not credited for determining when the leakage reaches the RWST (i.e., the leakage is assumed to reach the RWST instantaneously). Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis. Precedent for this assumption was previously established in the revised Shearon Harris Alternative Source Term submittal dated August 17, 2001 (specifically page 2.22-11) and the associated Shearon Harris Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 107 to NPF-63 issued October 12, 2001 (specifically page 31 of the Safety Evaluation Report).

The RWST pH equals 7.1 at the time of recirculation due to existing, highly concentrated sodium hydroxide from the spray additive tank (SAT) that is present in the mixing chamber in the RWST. Based upon the initial RWST pH of 7.1 at the start of recirculation, and based on information provided in NUREG-5950, it is expected that no elemental iodine will be regenerated in the RWST. However, for this analysis it was conservatively assumed that 1% of the particulate iodine would be converted to elemental iodine in the RWST. This conversion fraction is conservatively assumed to exist throughout the event even though the pH of the RWST would increase during the course of the event.

The elemental iodine generated in the RWST is assumed to become volatile and partition between the liquid and vapor space in the RWST based upon the temperature dependent partition coefficient for elemental iodine as presented in NUREG-5950. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWST since no boiling occurs in the RWST. The release of the activity

from the vapor space within the RWST is calculated based upon the displacement of air by the incoming leakage and the expansion due to the daily heating and cooling cycle of the contents (both air and liquid) of the RWST. The average daily temperature swing of 18.2 °F is applied for every 24-hour period for 30 days and no credit is taken for daily cooling. The final iodine release rate from the RWST is implemented via an adjustment to the leakage flow rate from the containment sump, which is applied to the entire iodine inventory in the containment sump, then released directly to the environment. The adjusted release rate is determined as follows:

$$Adjusted\ release\ rate = \left[\frac{(Leaked\ Volume \times Iodine\ Fraction)/RWT\ Liquid\ Volume}{Partition\ Coefficient\,(I_2)}\right] \times Air\ Flow\ Rate$$

where:

Iodine Fraction = 0.01 (Elemental iodine fraction available for release from the leaked water)

RWST Liquid Volume = Time dependent RWST liquid volume

Partition Coefficient (I_2) = Temperature dependent elemental iodine partition coefficient

Air Flow Rate = Time dependent air flow from RWST based on expansion and displacement

The adjusted release rate presented in Table 2.1-3 in then applied to the entire iodine inventory in the containment sump.

Containment purge is also assumed coincident with the beginning of the LOCA. Since the purge is isolated prior to the initial release of fission products from the core at 30 seconds, only the initial RCS activity (at an assumed 1.0 microcuries per gram DE I-131 and 100/E-bar gross activity) is available for release via this pathway. The release is modeled as an unfiltered release for 5 seconds until isolation occurs.

The release point for each of the above sources is presented in Table 2.1-1.

Transport Inputs

During the LOCA event, the activity collected by the secondary containment is assumed to be a filtered ground level release from the plant vent. The activity that bypasses the secondary containment is identified as being leaked via a ground level release from the containment without filtration. The activity from the ECCS leakage enters the secondary containment and is released to the environment via the plant vent after filtration. The activity from the RWST is modeled as an unfiltered ground level release from the RWST.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake during this mode is 1000 cfm of unfiltered fresh air.

- After the start of the event, the Control Room normal air intake is isolated due to a high containment pressure signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution was assumed to be 600 cfm of filtered makeup flow split equally through the two emergency intakes (using the X/Q for the worst intake with credit for dilution by the other intake), 150 cfm of unfiltered inleakage and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

LOCA Removal Inputs

Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.23 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region. No natural deposition removal of aerosols is credited in the sprayed regions. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 85.4% of the containment. Therefore, the Seabrook containment building atmosphere is not considered to be a single, well-mixed volume. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is 200 based on the maximum airborne elemental iodine concentration in the containment. The time for the containment sprays to reach an elemental iodine decontamination factor of 200 was determined by running a containment leakage case without environment leakage paths. Radioactive decay and natural deposition of iodine were conservatively left on as removal mechanisms contributing to the decontamination factor. Due to mixing between the sprayed and unsprayed regions of containment, the iodine activity in both containment regions was included in the determination of the time required to reach a decontamination factor of 200. The decontamination factor for elemental iodine reaches 200 at just over 2.92 hours.

The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 5.75 hr^{-1}, the time of a DF of 50 is computed with the same model used to determine the elemental iodine DF of 200. The time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.56 hours.

Filter removal in the Control Room Emergency Mode is simulated using conservative assumptions based on plant design data as listed in Table 1.6.3-1.

2.1.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

Three pathways for unfiltered inleakage to the control room were considered; inleakage via the diesel building, inleakage via the primary control room entrance (double air lock configuration), and inleakage via the emergency fire exit (two doors in series). A value of 10 cfm is typically assumed for door leakage for normal ingress/egress. However, this flow would be reduced or eliminated by a two-door vestibule. It was conservatively assumed that 20 cfm of total door leakage occurs via the most limiting door. The X/Qs for the fire exit are always more limiting than those for the primary control room entrance; therefore, all of the unfiltered inleakage via the doors was assumed to occur at the fire exit. For most release locations, the X/Qs for the fire exit are more limiting than the X/Qs for the diesel building inleakage. For these cases, the fire exit and diesel building intake were considered as separate paths for unfiltered inleakage. In cases where the diesel building is more limiting than the fire exit, all of the unfiltered inleakage was assumed to enter via the diesel building.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Table 1.8.2-1.

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. In addition, the MicroShield code, Version 5.05, Grove Engineering, is used to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis. It is qualified for this application and has been used to support licensing submittals that have been accepted by the NRC (for example, see Duane Arnold Energy Center submittal dated October 19, 2000 and associated NRC Safety Evaluation dated July

31, 2001.)

The post accident doses are the result of four distinct activity releases:

- Containment leakage.
- ESF system leakage into the CEVA.
- ESF system leakage into the RWST.
- Containment Purge at event initiation.

The dose to the Control Room occupants includes terms for:

1. Contamination of the Control Room atmosphere by intake and infiltration of radioactive material from the containment and ESF.
2. External radioactive plume shine contribution from the containment and ESF leakage releases. This term takes credit for Control Room structural shielding.
3. A direct shine dose contribution from the Containment's contained accident activity. This term takes credit for both Containment and Control Room structural shielding.
4. A direct shine dose contribution from the activity collected on the Control Room ventilation filters.

As shown in Table 2.1-5, the sum of the results of all four activity releases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.2 Fuel Handling Accident (FHA)

2.2.1 Background

This event consists of the drop of a single fuel assembly either in the Fuel Storage Building (FSB) or inside of Containment. The FHA is described in Section 15.7.4 of the UFSAR. The UFSAR description of the FHA specifies that all of the fuel rods in a single fuel assembly are damaged. In addition, a minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and FSB release locations.

This analysis bounds dropping a fuel assembly either inside the containment (with the personnel or equipment hatch open) or inside the FSB. Although filtration can be credited for the accident in the FSB there is sufficient margin to allow the analysis to be performed without crediting FSB filters. The source term released from the overlying water pool is the same for both the FSB and the containment cases. RG 1.183 imposes the same 2-hour criteria for the direct unfiltered release of the activity to the environment for either location. With the containment personnel hatch (a more limiting release point than the equipment hatch with respect to containment FHA releases) assumed open and filtration of the Fuel Storage Building exhaust not credited, the analyses are essentially identical for either the containment or the FSB release point except that the dispersion factors from the containment are slightly greater than the dispersion factors from the FSB.

To ensure that this analysis bounds the FHA in Containment or in the Fuel Storage Building, the most limiting combination of release point dispersion factors (*X/Q*) from the containment personnel hatch or the Fuel Storage Building release points is used. Use of the most limiting dispersion factors with no credit for FSB filtration assures the event results bound a Fuel Handling accident in either the containment or the Fuel Storage Building.

2.2.2 Compliance with RG 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," as discussed below:

1. Regulatory Position 1.1 - The amount of fuel damage is assumed to be all of the fuel rods in a single fuel assembly per UFSAR Section 15.7.4.3.

2. Regulatory Position 1.2 - The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of RG 1.183. Section 1.7 provides a discussion of how the FHA source term is developed. A listing of the FHA source term is provided in Table 1.7.5-1. The gap activity available for release is specified by Table 3 of RG 1.183. This activity is assumed to be released from the fuel instantaneously.

3. Regulatory Position 1.3 - The chemical form of radioiodine released from the damaged fuel into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The cesium iodide is assumed to completely dissociate in the spent fuel pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine.

4. Regulatory Position 2 - A minimum water depth of 23 feet is maintained above the damaged fuel assembly. Therefore, an overall decontamination factor of 200 is applied to the elemental and organic iodine based upon the composition specified in Regulatory Position 1.3.

5. Regulatory Position 3 - All of the noble gas released is assumed to exit the pool without mitigation.

All of the non-iodine particulate nuclides are assumed to be retained by the pool water.

6. Regulatory Position 4.1 - The analysis models the release from the FSB to the environment over a 2-hour period.

7. Regulatory Position 4.2 - No credit is taken for filtration of the release from the FSB.

8. Regulatory Position 4.3 - No credit is taken for dilution of the release in the FSB.

9. Regulatory Position 5.1 - The containment personnel hatch is assumed to be open at the time of the fuel handling accident.

10. Regulatory Position 5.2 - No automatic isolation of the containment is assumed for the FHA.

11. Regulatory Position 5.3 - The release from the containment fuel pool is assumed to leak to the environment over a two-hour period.

12. Regulatory Position 5.4 - No ESF filtration of the containment release is credited.

13. Regulatory Position 5.5 - No credit is taken for dilution or mixing in the containment atmosphere.

2.2.3 Methodology

The input assumptions used in the dose consequence analysis of the FHA are provided in Table 2.2-1. The limiting accident bounds a FHA inside of containment with the containment personnel hatch open or in the Fuel Storage Building without exhaust filtration. It is assumed that the fuel handling accident occurs at 80 hours after shutdown of the reactor per Licensing Submittal 02-06. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from a single fuel assembly into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. 100% of the noble gas released from the damaged fuel assembly is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the pool. The iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. A DF of 285 for elemental iodine and 1 for organic iodine is applied to the pool to accomplish the overall DF of 200 for the iodine release. The activity released from the pool is then assumed to leak to the environment over a two-hour period.

The FHA source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. Section 1.7 discusses the development of the FHA source term, which is listed in Table 1.7.5-1.

For this event, the Control Room ventilation system cycles through two modes of operation (the operational modes are summarized in Table 1.6.3-1):

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

2.2.4 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the location of the containment personnel hatch (bounds the FSB release) and the operational modes of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

The EAB and LPZ dose is determined using the *X/Q* factors provided in Table 1.8.2-1.

The radiological consequences of the FHA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.2-2 the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.3 Main Steamline Break (MSLB)

2.3.1 Background

This event consists of a double-ended break of one main steam line outside of containment. The radiological consequences of such an accident bound those of a MSLB inside containment. The affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cool down is achieved via the remaining unaffected SGs. This event is described in UFSAR Section 15.1.5.3.

2.3.2 Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1. Regulatory Position 1 – No fuel damage is postulated to occur for the Seabrook MSLB event.

2. Regulatory Position 2 – No fuel damage is postulated to occur for the Seabrook MSLB event. Two cases of iodine spiking are evaluated.

3. Regulatory Position 2.1 - One iodine spiking case assumes a reactor transient prior to the postulated MSLB that raises the primary coolant iodine concentration to the maximum allowed by Tech Specs, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.

4. Regulatory Position 2.2 - One case assumes the transient associated with the MSLB causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech Spec value of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 - Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by Tech Specs (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 500 gpd to the faulted SG and 940 gpd total to the other three SGs in order to maximize dose consequences.

8. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.

9. Regulatory Position 5.3 – The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 48 hours. The release of radioactivity from the unaffected SGs is assumed to continue until the steam release is terminated due to RHR initiation at 8 hours.

10. Regulatory Position 5.4 - All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

11. Regulatory Position 5.5.1 - In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected steam generators used for plant cooldown, tube bundle uncovery is not postulated; therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

12. Regulatory Position 5.5.2 – Tube bundle uncovery is not postulated for the unaffected SGs; therefore, this section does not apply. In the faulted SG, all of the fluid is assumed to flash and be released without mitigation.

13. Regulatory Position 5.5.3 - All leakage that does not immediately flash is assumed to mix with the bulk water.

14. Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%. No reduction in the release is assumed from the faulted SG.

15. Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for the intact SGs for Seabrook.

2.3.3 Other Assumptions

1. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the Tech Spec limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the intact SGs are provided in Table 2.3-2.

3. This evaluation assumes that the RCS mass remains constant throughout the MSLB event (no change in the RCS mass is assumed as a result of the MSLB or from the safety injection system).

4. The SG secondary side mass in the unaffected SGs is assumed to remain constant throughout the event.

5. Releases from the faulted main steam line (and associated SG) are postulated to occur from the main steam line associated with the most limiting atmospheric dispersion factors. Releases from the unaffected SGs are postulated to occur from the MSSV or ADV with the most limiting atmospheric dispersion factors.

2.3.4 Methodology

Input assumptions used in the dose consequence analysis of the MSLB are provided in Table 2.3-1. The postulated accident assumes a double-ended break of one main steam line outside containment. The radiological consequences of such an accident bound those of a MSLB inside of containment. Upon a MSLB, the affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cooldown is achieved via the remaining unaffected SGs.

The analysis assumes that the entire fluid inventory from the affected SG is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DE I-131 permitted by Tech Specs. Primary coolant is also released into the affected steam

generator by leakage across the SG tubes based on the Tech Spec primary to secondary leakage limits. Activity is released to the environment from the affected steam generator, as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the affected steam generator is completely isolated at 48 hours (primary system temperature less than 212°F). Additional activity, based on the Tech Spec primary-to-secondary leakage limits (SG tube leakage), is released via the unaffected SGs via steaming from the unaffected SGs MSSVs/ADVs for 8 hours (time of RHR initiation). These release assumptions are consistent with the requirements of RG 1.183.

Fuel damage is not postulated for the MSLB event per UFSAR section 15.1.5.3. Consistent with Regulatory Guide 1.183, Appendix E, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity released is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike; and (2) maximum accident-induced or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated MSLB event. The primary coolant iodine concentration is increased to the maximum value of 60 μCi/gm DE I-131 permitted by Technical Specification 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 2.3-3.

For the case of the accident-induced spike, the postulated MSLB event induces an iodine spike. The RCS activity is initially assumed to be 1.0 μCi/gm DE I-131 as allowed by Tech Specs. Iodine is released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 2.3-5.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- 20 cfm of unfiltered inleakage was assumed to enter the Control Room via the CR fire exit and 280 cfm was assumed to enter via the Diesel Building.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95 % for elemental and organic iodine.

2.3.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

Releases from the intact SGs are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs*. Releases from the faulted SG are assumed to occur from the location on a steam line that produces the most limiting *X/Qs*.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Table 1.8.2-1.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Cases for MSLB pre-accident and concurrent iodine spikes are analyzed. As shown in Table 2.3-6, the results of both cases for EAB dose, LPZ dose, and Control Room dose are within the appropriate regulatory acceptance criteria.

2.4 Steam Generator Tube Rupture (SGTR)

2.4.1 Background

This event is assumed to be caused by the instantaneous rupture of a Steam Generator tube that relieves to the lower pressure secondary system. No melt or clad breach is postulated for the Seabrook SGTR event. This event is described in UFSAR Section 15.6.3.3.

A single ASDV is assumed to stick open in the Seabrook SGTR analysis. Two stuck open ASDV scenarios are considered. Case 1 assumes that a single ASDV fails open when level reaches 33% in the affected SG. Case 2 assumes that a single ASDV fails open 3 minutes following reactor trip. The failed open ASDV is assumed to be reclosed 20 minutes after failing open.

2.4.2 Compliance with RG 1.183 Regulatory Positions

The SGTR dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1. Regulatory Position 1 - No fuel damage is postulated to occur for the Seabrook SGTR event.

2. Regulatory Position 2 - No fuel damage is postulated to occur for the Seabrook SGTR event. Two cases of iodine spiking are assumed.

3. Regulatory Position 2.1 - One case assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by Tech Specs, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.

4. Regulatory Position 2.2 - One case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech Spec limit of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 - Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by Tech Specs (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 313.33 gpd to the faulted SG and 1126.67 gpd total to the other three SGs in order to maximize dose consequences.

8. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.

9. Regulatory Position 5.3 - The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 48 hours. The release of radioactivity from the SGs is assumed to continue until shutdown cooling is in operation and steam release from the SGs is terminated (RHR initiation at 8 hours).

10. Regulatory Position 5.4 - The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP).

11. Regulatory Position 5.5 - All noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.

12. Regulatory Position 5.6 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:

   - Appendix E, Regulatory Position 5.5.1 – Tube uncovery is not postulated for this event; therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing for all steam generators.

   - Appendix E, Regulatory Position 5.5.2 - A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary. The portion that flashes immediately to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.

   - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage and ruptured tube flow that does not flash is assumed to mix with the bulk water.

   - Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

   - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

2.4.3 Other Assumptions

1. RCS and SG volume are assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events.

2. Data used to calculate the iodine equilibrium appearance rate are provided in Table 2.4-4, "Iodine Equilibrium Appearance Assumptions."

2.4.4 Methodology

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Table 2.4-1. This event is assumed to be caused by the instantaneous rupture of a steam generator tube releasing primary coolant to the lower pressure secondary system. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum. Valves in the condenser bypass lines would automatically close to protect the condenser thereby causing steam relief directly to the atmosphere from the ASDVs or MSSVs. This direct steam relief continues until it is terminated by initiation of RHR cooling (8 hours).

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the faulted SG relief valves. Additional activity, based

on the proposed primary-to-secondary leakage limits, is released via the MSSVs or ASDVs until the RHR system is placed in operation to continue heat removal from the primary system.

Per the Seabrook UFSAR, Section 15.6.3.3, no fuel melt or clad breach is postulated for the SGTR event. Consistent with RG 1.183 Appendix F, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike, and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration is increased to the maximum value of 60 µCi/gm DE I-131 permitted by Technical Specification 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 2.4-3. Primary coolant is released into the ruptured SG by the tube rupture and by a fraction of the total proposed allowable primary-to-secondary leakage. Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG ADVs. The unaffected SGs are used to cool down the plant during the SGTR event. Primary-to-secondary tube leakage is also postulated into the intact SGs. Activity is released via steaming from the SG MSSVs/ADVs until the decay heat generated in the reactor core can be removed by the RHR system at 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced spike, the postulated STGR event induces an iodine spike. The RCS activity is initially assumed to be 1.0 µCi/gm DE I-131 as allowed by Technical Specifications. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 2.4-5. All other release assumptions for this case are identical to those for the pre-accident spike case.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- 20 cfm of unfiltered inleakage was assumed to enter the Control Room via the CR fire exit and 280 cfm was assumed to enter via the Diesel Building.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95 % for elemental and organic iodine.

2.4.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

Releases from the intact and faulted SGs are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs* when combined with the limiting applicable control room intake.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q*

factors are provided in Table 1.8.2-1.

The radiological consequences of the SGTR Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-accident iodine spike and the accident-induced iodine spike, based on Tech Spec limits, are analyzed. In addition, two ASDV failure cases are analyzed. As shown in Table 2.4-6, the radiological consequences of the Seabrook SGTR event for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.5 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

2.5.1 Background

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the steam generator via the ASDVs and MSSVs. In addition, radioactivity is contained in the primary and secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident. This event is described in Section 15.3.4.3 of the UFSAR.

2.5.2 Compliance with RG 1.183 Regulatory Positions

The revised Locked Rotor dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 1.7.4-1. The inventory provided in Table 1.7.4-1 is then adjusted in the RADTRAD-NAI model for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of RG 1.183.

2. Regulatory Position 2 - Fuel damage is assumed for this event.

3. Regulatory Position 3 - Activity released from the damaged fuel is assumed to mix instantaneously and homogeneously throughout the primary coolant.

4. Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.

5. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the steam generators, as specified by Technical Specification 3.4.6.2, as 1 gpm total and 500 gallon per day to any one SG.

6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate Technical Specification.

7. Regulatory Position 5.3 - The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generators are terminated.

8. Regulatory Position 5.4 - The analysis assumes a coincident loss of offsite power.

9. Regulatory Position 5.5 - All noble gas radionuclides released from the primary system are assumed

released to the environment without reduction or mitigation.

10. Regulatory Position 5.6 - The steam generator tubes are assumed to remain covered throughout this event for Seabrook. Therefore, the iodine and transport model for release from the SGs is as follows:

    - Appendix E, Regulatory Position 5.5.1 – All four steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix instantaneously and homogeneously with the secondary water without flashing.

    - Appendix E, Regulatory Position 5.5.2 - None of the SG tube leakage is assumed to flash for this event.

    - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage is assumed to mix with the bulk water.

    - Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SG is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

2.5.3 Other Assumptions

1. RG 1.183, Section 3.6 - The assumed amount of fuel damage caused by the non-LOCA events is analyzed to determine the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and to determine the fraction of fuel elements for which fuel clad is breached. This analysis assumes DNB as the fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. For the Locked Rotor event, Table 3 of RG 1.183 specifies noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel.

2. The initial RCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131and 100/E-bar gross activity. The ratio of radioiodines to other radionuclides, provided in UFSAR Table 11.1-1, is assumed to be constant and the activities are scaled up to produce the TS limit of 1.0 μCi/gm Dose Equivalent I-131and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131. This analysis also conservatively assumes that the initial secondary coolant activity includes 10% of the primary coolant equilibrium concentration of alkali metals.

3. This analysis assumes that the DNB fuel damage is limited to 10% breached fuel assemblies. No fuel melt is assumed.

2.5.4 Methodology

Input assumptions used in the dose consequence analysis of the Locked Rotor event are provided in Table 2.5-1. This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will rise. The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial DNB margin and fuel damage.

For the purpose of this dose assessment, a total of 10% of the fuel assemblies are assumed damaged. A radial peaking factor of 1.65 is assumed. The activity released from the fuel gap is assumed to be released instantaneously and homogeneously through the primary coolant with source term from and release fractions per Appendix G of RG 1.183. Primary coolant is released to the SGs as a result of postulated primary-to-secondary leakage. Activity is released to the atmosphere via steaming from the steam generator ASDVs and MSSVs until the decay heat generated in the reactor core can be removed by the residual heat removal system 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the Control Room ventilation system cycles through three modes of operation (the operational modes are summarized in Table 1.6.3-1):

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

2.5.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the location of the release and the pathway for ingress into the CR. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

The EAB and LPZ dose consequences are determined using the *X/Q* factors provided in Table 1.8.2-1 for the appropriate time intervals.

The radiological consequences of the Locked Rotor event are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.5-3, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.6 Rod Cluster Control Assembly (RCCA) Ejection

2.6.1 Background

This event consists of the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. This event is the same as the Rod Ejection event referred to in RG 1.183. The RCCA Ejection results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is effected by steam release from the SG MSSVs/ASDVs. Two RCCA Ejection cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. The second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system. This event is described in the UFSAR, Section 15.4.8.3.

2.6.2 Compliance with RG 1.183 Regulatory Positions

The RCCA Ejection dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 1.7.4-1. The inventory provided in Table 1.7.4-1 is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in RG 1.183 Table 3 are adjusted to comply with the specific RG 1.183 Appendix H release requirements. For both the containment and secondary release cases, the activity available for release from the fuel gap for fuel that experiences DNB is assumed to be 10% of the noble gas and iodine inventory in the DNB fuel. For the containment release case for fuel that experiences fuel centerline melt (FCM), 100% of the noble gas and 25% of the iodine inventory in the melted fuel is assumed to be released to the containment. For the secondary release case for fuel that experiences FCM, 100% of the noble gas and 50% of the iodine inventory in the melted fuel is assumed to be released to the primary coolant.

2. Regulatory Position 2 - Fuel damage is assumed for this event.

3. Regulatory Position 3 - For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the SGs.

4. Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.

5. Regulatory Position 5 - The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.

6. Regulatory Position 6.1 - For the containment leakage case, natural deposition in the containment is credited. In addition, the Secondary Containment ventilation filtration is credited. Containment spray is not credited.

7. Regulatory Position 6.2 - The containment is assumed to leak at the TS maximum allowable rate of

0.15% for the first 24 hours and 0.075% for the remainder of the event.

8. Regulatory Position 7.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by proposed TS 3.4.6.2 (1.0 gpm total, 500 gallon per day to any one SG).

9. Regulatory Position 7.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.

10. Regulatory Position 7.3 - All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.

11. Regulatory Position 7.4 - Compliance with Appendix E Sections 5.5 and 5.6 is discussed below:

- Appendix E, Regulatory Position 5.5.1 – All four steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

- Appendix E, Regulatory Position 5.5.2 - None of the SG tube leakage is assumed to flash for this event.

- Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage is assumed to mix with the bulk water.

- Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

2.6.3 Other Assumptions

1. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131. This analysis also conservatively assumes that the initial secondary coolant activity includes 10% of the primary coolant equilibrium concentration of alkali metals from Table 1.7.2-1.

2. The steam mass release rates for the SGs are provided in Table 2.6-2.

3. This evaluation assumed that the RCS mass remains constant throughout the event.

4. The SG secondary side mass in the SGs is assumed to remain constant throughout the event.

5. Steam releases from the SGs are postulated to occur from the MSSV or ASDV with the most limiting atmospheric dispersion factors. For the RCCA Ejection inside of containment release case, releases are assumed to leak out of the containment via the same containment release points discussed for the LOCA in Section 2.1.

2.6.4 Methodology

Input assumptions used in the dose consequence analysis of the RCCA Ejection are provided in Table 2.6-

1. The postulated accident consists of two cases. One case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere, and the second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

For the containment release case, 100% of the activity is released instantaneously to the containment. The releases from the containment correspond to the same leakage points discussed for the LOCA in Section 2.1. Natural deposition of the released activity inside of containment is credited. In addition, the secondary containment building ventilation and filtration system is credited. Removal of activity via containment spray is not credited.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via steaming from the SG MSSVs/ASDVs until the decay heat generated in the reactor core can be removed by the Residual Heat Removal (RHR) system 8 hours into the event. Additional activity, based on the secondary coolant initial iodine concentration is assumed to be equal to the maximum value of 0.1 μCi/gm DE I-131 permitted by TS 3.7.1.4 (see Section 2.6.3 Assumption 1 for additional conservatism). Activity is released to the environment from the steam generator as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the steam generator steam release is terminated (at 8 hours for SDC initiation). These release assumptions are consistent with the requirements of RG 1.183.

The RCCA Ejection is evaluated with the assumption that 0.375% of the fuel experiences FCM and 15.0% of the fuel experiences DNB. The activity released from the damaged fuel corresponds to the requirements set out in Regulatory Position 1 of Appendix H to RG 1.183. A radial peaking factor of 1.65 is applied in the development of the source terms.

For this event, the Control Room ventilation system modes of operation are summarized in Table 1.6.3-1:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage assumed for the secondary side release or 190 cfm unfiltered inleakage assumed for the primary release.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage assumed for the secondary side release or 190 cfm unfiltered inleakage assumed for the primary release and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

2.6.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the pathway into the control room. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

For the RCCA secondary side release case, releases from the SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs*. For the RCCA Ejection containment release case, the *X/Qs* for containment leakage are assumed to be identical to those for the LOCA discussed in Section 2.1.

For the EAB and the LPZ dose the *X/Q* factors are provided in Table 1.8.2-1.

The radiological consequences of the RCCA Ejection are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.6-3, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.7 Failure of Small Lines Carrying Primary Coolant Outside of Containment

2.7.1 Background

This event is a rupture of a primary coolant letdown line outside of containment. Since RG 1.183 does not provide specific guidance to the analysis of this type of event, the general guidance of the Regulatory Guide will be supplemented with guidance of the Standard Review Plan (SRP) section 15.6.2 and consideration of the current licensing basis for this event. In accordance with SRP 15.6.2, the source term for this calculation will assume an accident-generated or concurrent iodine spike. In accordance with the assumptions of the current UFSAR section 15.6.2.3, a reactor trip is not predicted for this event. The dose assessment for this event is comprised of two separate release paths. Path 1 defines the leakage from the double ended rupture of the Letdown line in the Plant Auxiliary Building (PAB) outside of containment. Path 2 defines the release of activity through the secondary side steam release from the condenser.

2.7.2 Compliance with RG 1.183 Regulatory Positions

Since Regulatory Guide 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis. In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike in combination with the maximum leakage of primary fluid through the SG tubes into the secondary side. The RG 1.183 guidance provided for other events is applied to this event as applicable and appropriate.

The revised Letdown Line Rupture event dose consequence analysis is consistent with the guidance provided in RG 1.183, as discussed below:

1. Regulatory Position 2.2 of Appendix E - This guidance is used to define the concurrent iodine spike of 500 times the release rate corresponding to the iodine concentration at the equilibrium value (1.0 μCi/gm DE I-131).

2. Regulatory Position 3 of Appendix E - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

3. Regulatory Position 4 of Appendix E - The chemical form of radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the faulted SG and the unaffected SG to the environment (or containment) are assumed to be 97% elemental and 3% organic.

4. Regulatory Position 5.1 of Appendix E - The SGs are modeled as a single component with all SG tube leakage modeled into that component.

5. Regulatory Position 5.2 of Appendix E - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.

6. Regulatory Position 5.3 of Appendix E - Since a reactor trip is not predicted for this event, the primary-to-secondary leak rate is assumed to continue throughout the 30 day duration of the analysis.

7. Regulatory Position 5.4 of Appendix E - All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. All of the noble gas released from the primary system to the SGs is assumed to be released directly to the environment.

8. Regulatory Position 5.5.1 of Appendix E - For the steam generators, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

9. Regulatory Position 5.5.4 of Appendix E – It is conservatively assumed that the decontamination prescribed for the SGs in Regulatory Guide 1.183 is not applicable to the SGs under power operation. Therefore, no partition factor is applied to the activity as it is transferred from the SG to the turbine. Consistent with the pre-trip treatment of the secondary steam release during the current Steam Generator Tube Rupture at Seabrook, an iodine Decontamination Fraction of 99% will be assigned for the release from the condenser. It is similarly reasonable to assume that the 99% is equally applicable to all particulate released from the condenser. Therefore, the SG tube leakage will be modeled as a release from the RCS to the environment at the condenser location with a 99% filter efficiency for all particulates, and elemental and organic iodine.

10. Regulatory Position 5.6 of Appendix E - Steam generator tube bundle uncovery is not postulated for the SGs for Seabrook.

2.7.3 Other Assumptions

1. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. For a Letdown Line Rupture event outside of containment, releases from the faulted line are postulated to occur from the Primary Auxiliary Building at the location with the most limiting atmospheric dispersion factors. Releases from the secondary side are postulated to occur from the condenser.

2.7.4 Methodology

In accordance with the assumptions of the current UFSAR Section 15.6.2.3, the dose assessment for this event is comprised of two separate release paths. Path 1 defines the leakage from the double ended rupture of the letdown line in the Primary Auxiliary Building outside of containment with a direct release to the environment. Path 2 defines the release of RCS tube leakage through the secondary side via steam release through the condenser. Since RG 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis. In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike in combination with the maximum leakage of primary fluid through the SG tubes into the secondary side.

The accident generated appearance rate for the concurrent iodine spike is computed using the input in Table 2.7-2, with a 500 times multiplier on the normal appearance rate. The modeling of this spike is identical to that modeled for the MSLB concurrent spike case.

The Letdown Line Rupture flow rate is modeled as 140 gpm (at 62 lb_m/ft^3) for 30 minutes with a flashing fraction of 0.1815 as computed using the RG 1.183 guidance from position 5.4 of Appendix A for ECCS leakage for leakage at 380°F and 2235 psia. All of the noble gas in the letdown line rupture flow is released to the environment and the non-noble gas activity in the 0.1815 flashing fraction is assumed to be released (consistent with SRP 15.6.2 guidance).

For this event, the Control Room ventilation system modes of operation are summarized in Table 1.6.3-1:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the

Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

2.7.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the pathway into the control room. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

For the secondary side release case, releases from the SGs are assumed to occur from the condenser.

For the EAB and the LPZ dose the *X/Q* factors are provided in Table 1.8.2-1.

Reg. Guide 1.183 does not provide any direct guidance for the acceptance criteria for this event. However, the SRP states that the acceptance criteria is "a small fraction" of the 10CFR100 values which is further described as 10% of the limit. In applying the AST methodology to the letdown line break that same 10% interpretation is applied to the 10CFR part 50.67 limits for the LPZ and EAB dose. The acceptable dose limit for the Control Room (CR) is that specified in 10CFR50.67. For a Letdown Line Rupture, these limits are interpreted as:

Area	Dose Criteria	
EAB	2.5 rem TEDE	(for the worst two hour period)
LPZ	2.5 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the Letdown Line Rupture event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 2.7-3, "Dose Consequences," the radiological consequences of the Letdown Line Rupture event are all within the appropriate acceptance criteria.

2.8 Radioactive Gaseous Waste System Failure

2.8.1 Background

This event involves a major rupture of one of the Radioactive Gaseous Waste System (RGWS) components as currently presented in Section 15.7.1 of the Seabrook UFSAR. This analysis assumes that the ruptured RGWS component contains an inventory equivalent to the activity limit specified in Table 15.7-3 of the Seabrook UFSAR. The entire source term is applied to this RGWS component at the beginning of the event. The leak rate from the RGWS to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the activity to the environment within 2 hours. No hold-up, dilution or filtration of the tank release is assumed. The impact of the release is then computed as it disperses to the offsite receptors. The dose to Control Room operators is computed as the release is modeled to be treated by the Control Room Air Conditioning and Emergency Cleanup system during the 30-day period following the accident.

2.8.2 Compliance with RG 1.183 Regulatory Positions

RG 1.183 does not provide direct guidance relative to the Waste Gas system failures. Therefore, this analysis will rely primarily upon the current UFSAR licensing basis for guidance on performance of this event.

2.8.3 Methodology

The dose assessment model releases the above-prescribed inventory from the RGWS at a high rate of release. Per the existing analysis assumptions, the contents are released to the environment without any hold up, dilution or filtration over a 2 hour period.

For this event, the Control Room ventilation system cycles through two modes of operation (the operational modes are summarized in Table 1.6.3-1):

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

2.8.4 Radiological Consequences

The release-receptor point locations are chosen to model the distance from the release point to the Control Room intake. The *X/Q* values for the various combinations of release points and receptor locations are presented in Table 1.8.1-2. Table 1.8.1-3 presents the Release-Receptor pairs applicable to the Control Room dose from the RGWS release points for the different ingress pathways to the Control Room during the event.

For the EAB and LPZ dose analyses, the *X/Q* factors are provided in Table 1.8.2-1.

Reg. Guide 1.183 does not provide any requirement or dose limits for a RGWS failure; therefore, the acceptance criteria are set by the current Seabrook Licensing basis. Although Section 15.7.1.4 of the current Seabrook UFSAR concludes only that the consequences are "below the values specified in 10CFR Part 100", the RGWS failure will assume a more conventional criteria of 10% of the dose limit. This "small fraction" of the limit is consistent with the criteria established for the liquid waste system releases as stated in UFSAR Section 15.7.2.4. Therefore, the off-site dose acceptance criteria are established as 10% of the 10 CFR 50.67 limits. The control room dose limits are specified in 10CFR50.67. Therefore the dose limits are:

Area	Dose Criteria	
EAB	2.5 rem TEDE	(for the worst two hour period)
LPZ	2.5 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the RGWS failure event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 2.8-3, "RGWS Dose Consequences," the radiological consequences of the Radioactive Gaseous Waste System failure are all within the appropriate acceptance criteria.

2.9 Radioactive Liquid Waste System Failure

2.9.1 Background

This event involves a major rupture of one of the Radioactive Liquid Waste System (RLWS) components as currently presented in Section 15.7.2 of the Seabrook UFSAR. This analysis considers the two separate cases of the rupture of either the Boron Waste Storage Tank or the Letdown Degasifier. This analysis assumes that the ruptured RLWS component contains an inventory equivalent to the activity limit specified in Table 15.7-8 of the Seabrook UFSAR. The release activities are specified in Table 15.7-10 of the UFSAR. The entire release source term is applied to this RLWS component at the beginning of the event. The leak rate from the RLWS to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the activity to the environment within 2 hours. No hold-up, dilution or filtration of the tank release is assumed. The impact of the release is then computed as it disperses to the offsite receptors. The dose to Control Room operators is computed as the release is modeled to be treated by the Control Room Air Conditioning and Emergency Cleanup system during the 30-day period following the accident.

2.9.2 Compliance with RG 1.183 Regulatory Positions

RG 1.183 does not provide direct guidance relative to the Liquid Waste system failures. Therefore, this analysis will rely primarily upon the current UFSAR licensing basis for guidance on performance of this event.

2.9.3 Methodology

The dose assessment model releases the above-prescribed inventory from the RLWS at a high rate of release. Per the existing analysis assumptions, the contents are released to the environment without any hold up, dilution or filtration over a 2 hour period.

For this event, the Control Room ventilation system cycles through two modes of operation (the operational modes are summarized in Table 1.6.3-1):

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

2.9.4 Radiological Consequences

The release-receptor point locations are chosen to model the distance from the release point to the Control Room intake. The *X/Q* values for the various combinations of release points and receptor locations are presented in Table 1.8.1-2. Table 1.8.1-3 presents the Release-Receptor pairs applicable to the Control

Room dose from the RLWS release points for the different ingress pathways into the Control Room during the event.

For the EAB and LPZ dose analyses, the *X/Q* factors are provided in Table 1.8.2-1.

Reg. Guide 1.183 does not provide any requirement or dose limits for a RLWS failure; therefore, the acceptance criteria are set by the current Seabrook Licensing basis. Section 15.7.2.4 of the current Seabrook UFSAR concludes only that the consequences are within a "small fraction" of the values specified in 10CFR Part 100. Therefore, the off-site dose acceptance criteria are established as 10% of the 10 CFR 50.67 limits. The control room dose limits are specified in 10CFR50.67. Therefore the dose limits are:

Area	Dose Criteria
EAB	2.5 rem TEDE (for the worst two hour period)
LPZ	2.5 rem TEDE (for 30 days)
Control Room	5 rem TEDE (for 30 days)

The radiological consequences of the RLWS failure event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 2.9-3, "RLWS Dose Consequences," the radiological consequences of the Radioactive Liquid Waste System failure are all within the appropriate acceptance criteria.

2.10 Environmental Qualification (EQ)

The Seabrook Station UFSAR, Section 3.11, discusses equipment EQ due to the radiation environment. RG 1.183, Regulatory Position 6, allows the licensee to use either the AST or TID-14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. The Seabrook Station EQ analyses will continue to be based on TID-14844 assumptions.

3.0 Summary of Results

Results of the Seabrook Station radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered inleakage are summarized on Table 3-1.

4.0 Conclusion

Full implementation of the Alternative Source Term methodology, as defined in Regulatory Guide 1.183, into the design basis accident analyses has been made to support implementation of 10CFR50.67, to attempt to bound implementation of power uprate and to support control room habitability in the event of increases in control room unfiltered air inleakage. Analysis of the dose consequences of the Loss of Coolant Accident (LOCA), Fuel Handling Accident (FHA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Reactor Coolant Pump Shaft Seizure (Locked Rotor), Rod Cluster Control Assembly (RCCA) Ejection, Failure of Small Lines Carrying Primary Coolant Outside of Containment (Letdown Line Break), Radioactive Gaseous Waste System Failure and Radioactive Liquid Waste System Failure have been made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the Seabrook Station licensing basis and the calculated doses do not exceed the defined acceptance criteria.

This report supports a maximum allowable control room unfiltered air inleakage of *150* cfm.

5.0 References

5.1 Seabrook Station Updated Final Safety Analysis Report (UFSAR), Revision 8.

5.2 FPL Energy, Seabrook Station Unit No. 1 Technical Specifications (through Amendment 86).

5.3 TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

5.4 USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.

5.5 NEI 99-03, "Control Room Habitability Guidance," Nuclear Energy Institute, Revision 0 dated June 2001 and Revision 1 dated March 2003.

5.6 NRC Generic Letter 2003-01, "Control Room Habitability," June 12, 2003.

5.7 Code of Federal Regulations, 10CFR50.67, "Accident Source Term," revised 12/03/02.

5.8 NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

5.9 Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

5.10 Federal Guidance Report No. 12 (FGR 12), "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

5.11 Letter from J. Scarola to USNRC, "Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Revision to Alternate Source Term Methodology Analyses in Support of the Steam Generator Replacement and Power Uprate License Amendment Applications,: August 17, 2001.

5.12 Shearon Harris Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 107 to NPF-63 issued October 12, 2001.

5.13 Duane Arnold Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 240 to DPR-49 issued July 31, 2001.

5.14 ARCON96 Computer Code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664 and July 1997 errata).

5.15 MicroShield Version 5 "User's Manual "and "Verification & Validation Report, Rev. 5," Grove Engineering, both dated October 1996.

5.16 Oak Ridge National Laboratory, CCC-371, "RSICC Computer Code Collection – ORIGEN 2.1," May 1999.

5.17 "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations," NUREG/CR-2858, November 1982, (RSISS Computer Code Collection No. CCC-445).

5.18 Numerical Applications Inc., NAI-9912-04, Revision 3, "RADTRAD-NAI Version 1.1(QA) Documentation," April 2003.

5.19 Engineering Evaluation 03-010, "Alternative Source Term Design Inputs," Revision 1, FPL Energy Seabrook Station.

5.20 RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants," Rev. 1, February 1983.

5.21 Numerical Applications Inc., "Dose Methodology Quality Assurance Procedures," Revision 1, June 4, 2001.

5.22 NAI Calculation Number NAI-1101-002 Rev. 0, "Qualification of ORIGEN2.1 for Florida Power & Light AST Applications," August 8, 2002.

5.23 USNRC, Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

5.24 NUREG-0800, USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," September 1981 (or updates of specific sections).

5.25 Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983.

5.26 Indian Point Nuclear Generating Unit No. 2 Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 211 to DPR-26 issued July 27, 2000.

5.27 USNRC, Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 3, June 2001.

5.28 NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," June 3, 1999.

5.29 NRC Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," September 19, 1991.

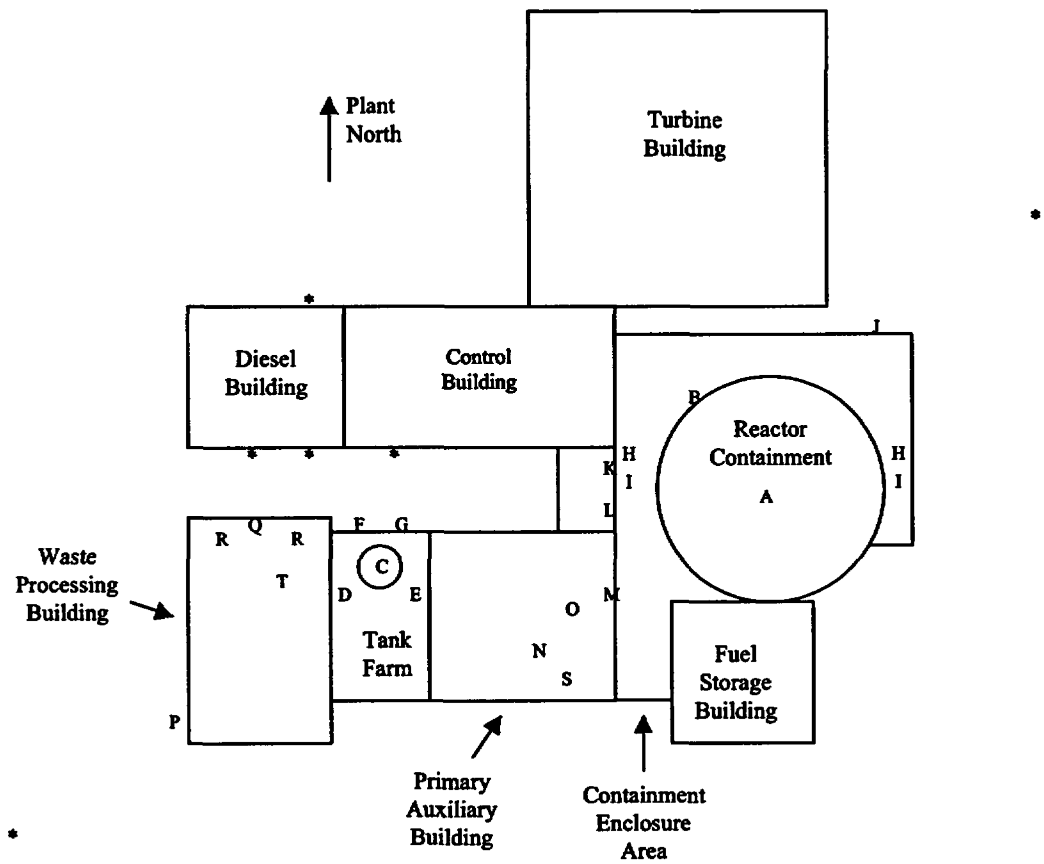
5.30 Letter from A. C. Bakken, III to USNRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Partial Response to Second Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability," January 15, 2002.

5.31 D. C. Cook Issuance of Amendments (IA) and Safety Evaluation (SE) for Amendment No. 271 to DPR-58 and Amendment No. 252 to DPR-74 issued November 14, 2002.

5.32 DBD-MET.1, Design Basis Document "Meteorological Monitoring System," Rev. 1.

5.33 2001 ASHRAE Fundamentals (IP Edition), American Society of Heating, Refrigerating, and Air-Conditioning Engineers, Inc.

Figure 1.8.1-1 Onsite Release-Receptor Location Sketch



(Not to scale)

* – Control Room Intakes / Receptor Point
A – Plant Vent
B – Personnel Hatch
C – RWST Tank
D – RWST Roof Vent WAH-FN-59A
E – RWST Roof Vent WAH-FN-59B
F – RWST Area Louver WAH-L-18
G – RWST Area Louver WAH-L-19
H – Closest MSSV
I – Closest ARV
J – Main Steam Line Release Point
K – Main Steam Line Chase (West) Panel (North)
L – Main Steam Line Chase (West) Panel (South)
M – PAB Louver PAH-L6A
N – PAB Louver PAH-L6D
O – PAB Fan PAH-FN46A
P – Waste Process Building SW Corner Roll-Up Door
Q – BWST Area Louver WAH-L-17A
R – BWST Tanks
S – Letdown Degassifier
T – Carbon Delay Bed (East)

Table 1.6.3-1 Control Room Ventilation System Parameters

Parameter	**Value**
Control Room Volume	246,000 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	1000 cfm
Unfiltered Inleakage (Total) LOCA Non-LOCA (except RCCA Ejection – containment release) RCCA Ejection – containment release	 150 cfm 300 cfm 190 cfm
Emergency Operation (single fan)	
Filtered Make-up Flow Rate	600 cfm
Filtered Recirculation Flow Rate	390 cfm (entire 10% tolerance on total CR filter flow conservatively applied to reduce recirculation flow)
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage (Total) LOCA Non-LOCA (except RCCA Ejection – containment release) RCCA Ejection – containment release	 150 cfm 300 cfm 190 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%

Table 1.6.3-2 LOCA Direct Shine Dose

Source	Direct Shine Dose (rem)
Containment	0.001
Filters	0.429
External Cloud	0.020
Total	**0.450**

Table 1.7.2-1 Primary Coolant Source Term

Nuclide	µCi/gm	Nuclide	µCi/gm
I-131	0.7727	SR-91	1.341E-01
I-132	0.2813	Y-90	9.515E-04
I-133	1.2363	Y-91	2.509E-02
I-134	0.1793	Y-92	4.325E-03
I-135	0.6800	ZR-95	2.898E-03
H-3	2.163E+01	NB-95	2.941E-03
KR-83M	1.860E+00	MO-99	1.427E+01
KR-85M	7.353E+00	CS-134	1.903E+00
KR-85	5.623E-01	CS-136	9.515E-01
KR-87	5.623E+00	CS-137	9.515E+00
KR-88	1.471E+01	BA-140	1.946E-02
XE-131M	2.898E-01	LA-140	6.055E-03
XE-133M	2.465E+00	CE-144	1.903E-03
XE-133	1.081E+02	MN-54	1.341E-03
XE-135M	3.547E+00	CO-58	6.920E-02
XE-135	1.341E+01	CO-60	8.650E-03
XE-138	3.071E+00	FE-59	4.325E-03
SR-89	1.773E-02	CR-51	8.218E-03
SR-90	7.785E-04	FE-55	6.920E-03

Table 1.7.3-1 Secondary Side Source Term

Isotope	µCi/gm
I-131	0.07727
I-132	0.02813
I-133	0.12363
I-134	0.01793
I-135	0.06800

Table 1.7.4-1 LOCA Containment Leakage Source Term

Nuclide	Curies	Nuclide	Curies
Kr-85	1.263E+06	Pu-239	3.991E+04
Kr-85m	2.492E+07	Pu-240	7.068E+04
Kr-87	4.765E+07	Pu-241	1.865E+07
Kr-88	6.703E+07	Am-241	1.847E+04
Rb-86	3.028E+05	Cm-242	8.770E+06
Sr-89	9.245E+07	Cm-244	2.598E+06
Sr-90	1.002E+07	I-130	7.585E+06
Sr-91	1.134E+08	Kr-83m	1.186E+07
Sr-92	1.231E+08	Xe-138	1.609E+08
Y-90	1.048E+07	Xe-131m	1.179E+06
Y-91	1.198E+08	Xe-133m	6.446E+06
Y-92	1.237E+08	Xe-135m	4.221E+07
Y-93	1.434E+08	Cs-138	1.787E+08
Zr-95	1.644E+08	Cs-134m	8.436E+06
Zr-97	1.613E+08	Rb-88	6.819E+07
Nb-95	1.664E+08	Rb-89	8.726E+07
Mo-99	1.892E+08	Sb-124	3.729E+05
Tc-99m	1.657E+08	Sb-125	2.414E+06
Ru-103	1.879E+08	Sb-126	2.109E+05
Ru-105	1.510E+08	Te-131	9.249E+07
Ru-106	1.000E+08	Te-133	1.184E+08
Rh-105	1.344E+08	Te-134	1.609E+08
Sb-127	1.390E+07	Te-125m	5.259E+05
Sb-129	3.781E+07	Te-133m	7.129E+07
Te-127	1.381E+07	Ba-141	1.585E+08
Te-127m	1.874E+06	Ba-137m	1.294E+07
Te-129	3.719E+07	Pd-109	6.296E+07
Te-129m	5.518E+06	Rh-106	1.115E+08
Te-131m	1.598E+07	Rh-103m	1.692E+08
Te-132	1.460E+08	Tc-101	1.764E+08
I-131	1.051E+08	Eu-154	2.003E+06
I-132	1.491E+08	Eu-155	1.387E+06
I-133	1.988E+08	Eu-156	4.767E+07
I-134	2.152E+08	La-143	1.466E+08
I-135	1.872E+08	Nb-97	1.628E+08
Xe-133	1.994E+08	Nb-95m	1.177E+06
Xe-135	5.012E+07	Pm-147	1.379E+07
Cs-134	3.258E+07	Pm-148	2.968E+07
Cs-136	8.347E+06	Pm-149	6.826E+07
Cs-137	1.365E+07	Pm-151	2.409E+07
Ba-139	1.747E+08	Pm-148m	3.426E+06
Ba-140	1.684E+08	Pr-144	1.350E+08
La-140	1.750E+08	Pr-144m	1.609E+06

Nuclide	Curies	Nuclide	Curies
La-141	1.593E+08	**Sm-153**	7.818E+07
La-142	1.538E+08	**Y-94**	1.448E+08
Ce-141	1.623E+08	**Y-95**	1.560E+08
Ce-143	1.476E+08	**Y-91m**	6.581E+07
Ce-144	1.340E+08	**Br-82**	8.712E+05
Pr-143	1.464E+08	**Br-83**	1.183E+07
Nd-147	6.392E+07	**Br-84**	2.044E+07
Np-239	2.922E+09	**Am-242**	1.285E+07
Pu-238	5.151E+05	**Np-238**	7.062E+07
		Pu-243	1.266E+08

Table 1.7.5-1 Fuel Handling Accident Source Term

Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)
Kr-85	1.080E+04	Xe-133	1.705E+06	Sb-126	1.803E+03
Kr-85m	2.130E+05	Xe-135	4.285E+05	Te-131	7.907E+05
Kr-87	4.074E+05	Cs-134	2.785E+05	Te-133	1.012E+06
Kr-88	5.731E+05	Cs-136	7.136E+04	Te-134	1.376E+06
Rb-86	2.589E+03	Cs-137	1.167E+05	Te-125m	4.496E+03
Sr-89	7.904E+05	Ba-139	1.494E+06	Te-133m	6.095E+05
Sr-90	8.566E+04	Ba-140	1.440E+06	Ba-141	1.355E+06
Sr-91	9.695E+05	La-140	1.496E+06	Ba-137m	1.106E+05
Sr-92	1.052E+06	La-141	1.362E+06	Pd-109	5.383E+05
Y-90	8.960E+04	La-142	1.315E+06	Rh-106	9.532E+05
Y-91	1.024E+06	Ce-141	1.388E+06	Rh-103m	1.447E+06
Y-92	1.058E+06	Ce-143	1.262E+06	Tc-101	1.508E+06
Y-93	1.226E+06	Ce-144	1.146E+06	Eu-154	1.712E+04
Zr-95	1.405E+06	Pr-143	1.252E+06	Eu-155	1.186E+04
Zr-97	1.379E+06	Nd-147	5.465E+05	Eu-156	4.075E+05
Nb-95	1.423E+06	Np-239	2.498E+07	La-143	1.253E+06
Mo-99	1.618E+06	Pu-238	5.425E+03	Nb-97	1.392E+06
Tc-99m	1.417E+06	Pu-239	3.412E+02	Nb-95m	1.006E+04
Ru-103	1.606E+06	Pu-240	6.043E+02	Pm-147	1.179E+05
Ru-105	1.291E+06	Pu-241	1.594E+05	Pm-148	2.537E+05
Ru-106	8.549E+05	Am-241	1.579E+02	Pm-149	5.836E+05
Rh-105	1.149E+06	Cm-242	7.498E+04	Pm-151	2.060E+05
Sb-127	1.188E+05	Cm-244	3.232E+04	Pm-148m	2.929E+04
Sb-129	3.232E+05	I-130	6.485E+04	Pr-144	1.154E+06
Te-127	1.181E+05	Kr-83m	1.014E+05	Pr-144m	1.376E+04
Te-127m	1.602E+04	Xe-138	1.376E+06	Sm-153	6.684E+05
Te-129	3.179E+05	Xe-131m	1.008E+04	Y-94	1.238E+06
Te-129m	4.717E+04	Xe-133m	5.511E+04	Y-95	1.334E+06
Te-131m	1.366E+05	Xe-135m	3.609E+05	Y-91m	5.626E+05
Te-132	1.248E+06	Cs-138	1.528E+06	Br-82	7.448E+03
I-131	8.985E+05	Cs-134m	7.212E+04	Br-83	1.011E+05
I-132	1.275E+06	Rb-88	5.830E+05	Br-84	1.747E+05
I-133	1.700E+06	Rb-89	7.460E+05	Am-242	1.099E+05
I-134	1.840E+06	Sb-124	3.188E+03	Np-238	6.347E+05
I-135	1.600E+06	Sb-125	2.064E+04	Pu-243	1.082E+06

Table 1.8.1-1 Release-Receptor Combination Parameters for Analysis Events

Release Point	Receptor Point	Release Height (ft)	Release Height (m)	Receptor Height (ft)	Receptor Height (m)	Distance (ft)	Distance (m)	Direction with respect to true north
Plant Vent	East Intake	185	56.4	6.5	2.0	352.34	107.3	196
Plant Vent	CR Fire Exit Door	185	56.4	5	1.5	215.31	65.6	67
Plant Vent	Diesel Building Intake	185	56.4	28.5	8.7	246.52	75.1	65
Closest Containment Surface Point	East Intake	6.5	2.0	6.5	2.0	272.09	82.9	196
Closest Containment Surface Point	CR Fire Exit Door	5	1.5	5	1.5	135.06	41.1	67
Closest Containment Surface Point	Diesel Building Intake	28.5	8.7	28.5	8.7	166.27	50.6	65
RWST	West Intake	50	15.2	8.25	2.5	315.3	96.1	7
RWST	CR Fire Exit Door	50	15.2	5	1.5	75.54	23.0	151
RWST	Diesel Building Intake	50	15.2	28.5	8.7	77.97	23.7	127
Containment Personnel Hatch	East Intake	9.5	2.9	6.5	2.0	372.69	113.5	210
Containment Personnel Hatch	CR Fire Exit Door	9.5	2.9	5	1.5	149.95	45.7	49
Containment Personnel Hatch	Diesel Building Intake	9.5	2.9	28.5	8.7	181.88	55.4	50
Main Steam Line Closest Point	East Intake	20.58	6.3	6.5	2.0	202.5	61.7	210
Main Steam Line Chase (West) Panel (North)	CR Fire Exit Door	38.38	11.7	5	1.5	112.26	34.2	55

Release Point	Receptor Point	Release Height (ft)	Release Height (m)	Receptor Height (ft)	Receptor Height (m)	Distance (ft)	Distance (m)	Direction with respect to true north
Main Steam Line Chase (West) Panel (North)	Diesel Building Intake	38.38	11.7	28.5	8.7	144.26	43.9	55
Closest MSSV	East Intake	53.16	16.2	6.5	2.0	251.94	76.7	191
Closest ARV	East Intake	54.5	16.6	6.5	2.0	282.71	86.1	185
Closest MSSV	CR Fire Exit Door	53.16	16.2	5	1.5	115.6	35.2	57
Closest ARV	CR Fire Exit Door	54.5	16.6	5	1.5	125.91	38.3	75
Closest MSSV	Diesel Building Intake	53.16	16.2	28.5	8.7	147.55	44.9	56
Closest ARV	Diesel Building Intake	54.5	16.6	28.5	8.7	156.05	47.5	71
Primary Auxiliary Building Louver PAH-L6D	West Intake	61	18.6	8.25	2.5	331.74	101.1	22
Primary Auxiliary Building Fan PAH-FN46A	CR Fire Exit Door	88	26.8	5	1.5	122.85	37.4	101
Primary Auxiliary Building Fan PAH-FN46A	Diesel Building Intake	88	26.8	28.5	8.7	146.41	44.6	92
Turbine Building Closest Point	East Intake	6.5	2.0	6.5	2.0	211.77	64.5	234
Turbine Building Closest Point	CR Fire Exit Door	5	1.5	5	1.5	117.6	35.8	1
Turbine Building Closest Point	Diesel Building Intake	28.5	8.7	28.5	8.7	102	31.0	54
Waste Process Building SW Corner Roll-Up Door	West Intake	8.25	2.5	8.25	2.5	164.03	49.9	4

Release Point	Receptor Point	Release Height (ft)	Release Height (m)	Receptor Height (ft)	Receptor Height (m)	Distance (ft)	Distance (m)	Direction with respect to true north
Carbon Delay Bed (East)	Diesel Building Intake	41.42	12.6	28.5	8.7	80.5	24.5	144
BWST (West)	Diesel Building Intake	22.67	6.9	28.5	8.7	53.67	16.3	144

Notes:

1. Release heights are calculated as 20 feet less than the reference elevations to account for the plant grade elevation.

2. The closest/limiting MSSV is MSSV-V40 for the East Intake and MSSV-V-54 for the control room fire exit door and Diesel building intakes. The closest/limiting ARV is off of main steam line MS-4002 for the East Intake and main steam line MS-4003 for the control room fire exit door and Diesel building intakes.

3. Release and receptor points are considered to be at the centerpoint or centerline of all openings.

4. The closest main steam line break point for the East Intake is off of main steam line MS-4002.

Table 1.8.1-2 Onsite Atmospheric Dispersion (*X/Q*) Factors for Analysis Events

This table summarizes the results for *X/Q* factors for the control room intakes for the various accident scenarios. Values are presented for the release point to the unfavorable control room makeup air intake and the unfiltered inleakage point, which is a Diesel Building intake and/or the control room fire exit door. These values are not corrected for Control Room Occupancy Factors but the control room makeup air intakes do include taking credit for dilution. Based on the layout of the site and the fact that both makeup air intakes have equal flow rates, the base χ/Q values may be reduced by a factor of 2. These reduced values are listed in the table below.

Some of the event analyses take credit for a factor of 5 reduction on the base χ/Q values to account for buoyant plume rise from the MSSVs and ADVs in accordance with Section 6 of Regulatory Guide 1.194. This reduction factor is not reflected in the table below. Note that the letters that indicate the release-receptor pairs do not necessarily correspond with the release identification letters on Figure 1.8.1-1.

Release-Receptor Pair	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
A	Plant Vent	East Intake	2.34E-04	1.85E-04	6.75E-05	4.62E-05	3.87E-05
B	Plant Vent	CR Fire Exit Door	7.54E-04	5.03E-04	2.00E-04	1.45E-04	9.89E-05
C	Plant Vent	Diesel Building Intake	7.01E-04	4.74E-04	1.89E-04	1.37E-04	8.97E-05
D	Closest Containment Surface Point	East Intake	4.40E-04	3.46E-04	1.29E-04	8.40E-05	6.80E-05
E	Closest Containment Surface Point	CR Fire Exit Door	3.08E-03	2.17E-03	8.48E-04	6.31E-04	4.64E-04
F	Closest Containment Surface Point	Diesel Building Intake	2.06E-03	1.48E-03	5.79E-04	4.29E-04	3.11E-04
G	RWST	West Intake	3.54E-04	2.75E-04	9.70E-05	6.90E-05	4.37E-05
H	RWST	CR Fire Exit Door	7.52E-03	3.85E-03	1.26E-03	9.29E-04	7.23E-04
I	RWST	Diesel Building Intake	5.06E-03	2.85E-03	9.00E-04	7.17E-04	6.17E-04
J	Containment Personnel Hatch	East Intake	2.84E-04	2.48E-04	1.04E-04	6.50E-05	5.10E-05
K	Containment Personnel Hatch	CR Fire Exit Door	2.84E-03	2.30E-03	8.67E-04	5.87E-04	3.70E-04
L	Containment Personnel Hatch	Diesel Building Intake	1.97E-03	1.60E-03	5.99E-04	4.04E-04	2.58E-04

Release-Receptor Pair	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
M	Main Steam Line Closest Point	East Intake	8.70E-04	7.85E-04	3.22E-04	2.02E-04	1.61E-04
N	Main Steam Line Chase (West) Panel (North)	CR Fire Exit Door	4.55E-03	3.72E-03	1.38E-03	9.67E-04	6.35E-04
O	Main Steam Line Chase (West) Panel (North)	Diesel Building Intake	3.11E-03	2.50E-03	9.37E-04	6.53E-04	4.29E-04
P	Closest MSSV	East Intake	5.45E-04	4.50E-04	1.56E-04	9.85E-05	8.00E-05
Q	Closest ARV	East Intake	4.44E-04	3.38E-04	1.16E-04	7.30E-05	6.05E-05
R	Closest MSSV	CR Fire Exit Door	4.11E-03	3.31E-03	1.24E-03	8.72E-04	5.86E-04
S	Closest ARV	CR Fire Exit Door	3.49E-03	2.79E-03	1.02E-03	7.54E-04	5.45E-04
T	Closest MSSV	Diesel Building Intake	2.89E-03	2.39E-03	8.87E-04	6.17E-04	4.11E-04
U	Closest ARV	Diesel Building Intake	2.64E-03	2.11E-03	7.82E-04	5.71E-04	4.07E-04
V	Primary Auxiliary Building Louver PAH-L6D	West Intake	3.21E-04	2.68E-04	1.02E-04	6.75E-05	3.72E-05
W	Primary Auxiliary Building Fan PAH-FN46A	CR Fire Exit Door	2.91E-03	1.98E-03	6.61E-04	5.09E-04	4.37E-04
X	Primary Auxiliary Building Fan PAH-FN46A	Diesel Building Intake	2.63E-03	1.81E-03	6.48E-04	4.86E-04	3.95E-04
Y	Turbine Building Closest Point	East Intake	8.40E-04	7.65E-04	3.44E-04	2.41E-04	1.91E-04
Z	Turbine Building Closest Point	CR Fire Exit Door	4.49E-03	3.22E-03	1.19E-03	8.27E-04	5.99E-04
AA	Turbine Building Closest Point	Diesel Building Intake	5.95E-03	4.80E-03	1.79E-03	1.24E-03	8.00E-04

Release-Receptor Pair	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
BB	Waste Process Building SW Corner Roll-Up Door	West Intake	1.18E-03	8.85E-04	3.25E-04	2.28E-04	1.47E-04
CC	Carbon Delay Bed (East)	Diesel Building Intake	8.57E-03	4.46E-03	1.43E-03	1.11E-03	8.37E-04
DD	BWST (West)	Diesel Building Intake	1.86E-02	9.65E-03	3.08E-03	2.39E-03	1.84E-03

Table 1.8.1-3 Release-Receptor Point Pairs Assumed for Analysis Events

Event	Filtered Makeup	Unfiltered Inleakage Through Diesel Building	Unfiltered Inleakage Through CR Fire Exit Door
LOCA			
- Containment Leakage CEVA Release	A	C	B
- Containment Leakage CEVA Bypass	D	F	E
- ECCS Leakage CEVA Release	A	C	B
- ECCS Leakage CEVA Bypass	D	F	E
- RWST Backleakage	G	I	H
- Containment Purge	A	N/A	B
FHA (bounding for Containment and FSB)	J	L	K
MSLB			
- Break Release	M	O	N
- MSSV/ASDV Release	P (prior to 2.5 hours, also applies plume rise factor of 5 reduction) Q (after 2.5 hours)	T (prior to 2.5 hours, also applies plume rise factor of 5 reduction) U (after 2.5 hours)	R (prior to 2.5 hours, also applies plume rise factor of 5 reduction) S (after 2.5 hours)
SGTR	P (prior to 2.5 hours for the iodine spike – also applies plume rise factor of 5 reduction; entire transient for the noble gas release) Q (after 2.5 hours for the iodine spike)	T (prior to 2.5 hours for the iodine spike – also applies plume rise factor of 5 reduction; entire transient for the noble gas release) U (after 2.5 hours for the iodine spike)	R (prior to 2.5 hours for the iodine spike – also applies plume rise factor of 5 reduction; entire transient for the noble gas release) S (after 2.5 hours for the iodine spike)
Locked Rotor	P (prior to 2.5 hours, also applies plume rise factor of 5 reduction) Q (after 2.5 hours)	T (prior to 2.5 hours, also applies plume rise factor of 5 reduction) U (after 2.5 hours)	R (prior to 2.5 hours, also applies plume rise factor of 5 reduction) S (after 2.5 hours)
RCCA Ejection			
- Containment Leakage CEVA Release	A	C	B
- Containment Leakage CEVA Bypass	D	F	E
- Secondary Side Release	P (prior to 2.5 hours, also applies plume rise factor of 5 reduction) Q (after 2.5 hours)	T (prior to 2.5 hours, also applies plume rise factor of 5 reduction) U (after 2.5 hours)	R (prior to 2.5 hours, also applies plume rise factor of 5 reduction) S (after 2.5 hours)

Event	Filtered Makeup	Unfiltered Inleakage Through Diesel Building	Unfiltered Inleakage Through CR Fire Exit Door
Small Line Break			
- Break Release	V	X	W
- Condenser Release	Y	AA	Z
Radioactive Gaseous Waste System Failure	BB	CC	N/A*
Radioactive Liquid Waste System Failure	BB	DD	N/A*

* It is conservative for these release points to assume that all unfiltered Control Room inleakage is through the Diesel Building (i.e., no unfiltered inleakage through the Control Room Fire Exit door).

Table 1.8.2-1 Offsite Atmospheric Dispersion (*X/Q*) Factors for Analysis Events

Time Period	EAB *X/Q* (sec/m^3)	LPZ *X/Q* (sec/m^3)
0-2 hours	3.17E-04	1.54E-04
0-8 hours	2.08E-04	8.63E-05
8-24 hours	1.68E-04	6.46E-05
1-4 days	1.06E-04	3.45E-05
4-30 days	5.51E-05	1.40E-05

The above table summarizes the maximum *X/Q* factors for the EAB and LPZ.

Table 2.1-1 Loss of Coolant Accident (LOCA) – Inputs and Assumptions

Input/Assumption	Value
Release Inputs:	
Core Power Level	3659 MW_{th} (include uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
RCS Mass	539,037 lb_m
Core Fission Product Inventory	Table 1.7.4-1
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.15% (by weight)/day 0.075% (by weight)/day
LOCA release phase timing and duration	Table 2.1-2
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	RG 1.183, Sections 3.1, 3.2, and Table 2
<u>ECCS Systems Leakage (from 26 minutes to 30 days)</u> Sump Volume (minimum) ECCS Leakage (2 times allowed value) Flashing Fraction Chemical form of the iodine in the sump water at the time of recirculation (based on pH history) after 26 minutes Released via plant vent after filtration	 69,159.75 $ft.^3$ 48 gpd All available elemental and organic iodine assumed to be released 98.85% aerosol, 1.0% elemental, and 0.15% organic

Input/Assumption	Value
RWST Back-leakage	
Sump Volume (minimum)	69,159.75 $ft.^3$
ECCS Leakage to RWST (2 times allowed value)	0.9595 gpm
Flashing Fraction (elemental iodine assumed to be released into tank air space based upon partition factor)	0 % assumed
RWST liquid/vapor elemental iodine partition factor	See Table 2.1-4
Chemical form of iodine in the RWST (based on Sump and RWST pH history)	99% aerosol, 1.0% elemental
Initial RWST Liquid Inventory (minimum at time of recirculation)	47,000 gallons
Release from RWST Vapor Space	Table 2.1-3
Containment Purge Release (unfiltered)	1,000 cfm for 5 seconds
Removal Inputs:	
Containment Aerosol/Particulate Natural Deposition (only credited in unsprayed regions)	0.1/hour
Containment Elemental Iodine Wall Deposition	2.23/hour
Containment Sprayed Region Volume	2,309,000 ft^3
Containment Unsprayed Region Volume	395,000 ft^3
Flowrate Sprayed and Unsprayed Volumes (Based on two turnovers per hour of unsprayed volume)	13,000 cfm
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Particulate Iodine Time to reach DF of 50	 20/hour 2.92 hours 5.75/hour 3.56 hours
Spray Initiation Time	65 seconds (0.018 hours)
Control Room Ventilation System Time of automatic control room normal intake isolation and switch to emergency mode	Table 1.6.3-1 30 seconds
Containment enclosure emergency air cleaning system filter efficiency	Particulate – 95% Elemental – 95% Organic – 85%
Containment enclosure emergency air cleaning system drawdown time	4.5 minutes

Input/Assumption	Value
Containment enclosure emergency air cleaning system bypass fraction	60%
Containment Purge Filtration	0 %
Transport Inputs:	
Containment Leakage Release	Plant vent (filtered by CEVA) and closest containment point (CEVA bypass)
ECCS Leakage	Plant vent
RWST Backleakage	RWST tank
Containment Purge	Plant vent
Personnel Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Breathing Rates Offsite Control Room	RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.1-2 LOCA Release Phases

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

* From RG 1.183, Table 4

Table 2.1-3 Adjusted Release Rate from RWST

Time (hours)	Release Rate (cfm)
0	1.0403E-05
22	2.7185E-05
24	6.4831E-05
100	1.0186E-04
200	1.3059E-04
300	1.5290E-04
400	1.7027E-04
500	1.8411E-04
600	1.8538E-04
700	1.8044E-04

Table 2.1-4 RWST Elemental Iodine Partition Factor

Time (hours)	Partition Factor
0	47.42
22	45.34
24	45.21
100	42.30
200	41.12
300	40.74
400	40.70
500	40.82
600	40.92
700	41.01

Table 2.1-5 LOCA Dose Consequences

Dose Component	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
Containment Purge	4.2391E-04	2.0602E-04	3.8398E-04
Containment Leakage	4.3932	3.2278	3.7956
ECCS Leakage	9.6473E-04	6.4984E-03	8.8367E-03
RWST Backleakage	3.3815E-03	0.14140	0.47062
Radiation Shine			0.45
Total	4.40	3.38	4.73
Acceptance Criteria	25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 150 cfm

Table 2.2-1 Fuel Handling Accident (FHA) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	3659 MW_{th} (including uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Discharged Fuel Assembly Burnup	45,000 – 62,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
Number of Fuel Assemblies in the Core	193
Number of Fuel Assemblies Damaged	1
Delay Before Spent Fuel Movement	80 hours
FHA Source Term for a Single Assembly	Table 1.7.5-1
Water Level Above Damaged Fuel Assembly	23 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic – 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Duration of Release to Environment	2 hrs
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Time of Control Room Ventilation System Isolation	30 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.2-2 Fuel Handling Accident Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)*
FHA	1.41	0.69	2.39
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.3-1 Main Steam Line Break (MSLB) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3659 MW_{th} (includes uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 1.7.3-1
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	Faulted SG – 500 gallons/day Intact SGs – 940 gallons/day
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	48 hours
RCS Mass	539,037 lb_m
SG Secondary Side Mass	Maximum (Hot Zero Power) – 166,000 lb_m (used for faulted SG to maximize release) Minimum (Hot Full Power) – 99,304 lb_m per SG for a total of 297,912 lb_m (used for intact SGs to maximize concentration)
Release from Faulted SG	Instantaneous
Steam Release from Intact SGs	Table 2.3-2
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of automatic control room normal intake isolation and switch to emergency mode	Table 1.6.3-1 30 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183 Section 4.2.6

Table 2.3-2 Intact SGs Steam Release Rate

Time (hours)	Intact SGs Steam Release Rate* (lb_m/min)
0 – 2	3383.3
2 – 8	2563.9
8 – 720.0	0

*Total release rate for all three (3) intact SGs.

Table 2.3-3 60 μCi/gm D.E. I-131 Activities

Isotope	Activity (μCi/gm)
Iodine-131	46.36
Iodine-132	16.88
Iodine-133	74.18
Iodine-134	10.76
Iodine-135	40.80

Table 2.3-4 Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow *	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus 10% uncertainty

Table 2.3-5 Concurrent Iodine Spike (500 x) Activity Appearance Rate

Isotope	Appearance Rate (Ci/min)
Iodine-131	209.029344
Iodine-132	235.958907
Iodine-133	404.536383
Iodine-134	317.823719
Iodine-135	315.402448

Table 2.3-6 MSLB Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)*
MSLB pre-accident iodine spike	0.07	0.11	1.11
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
MSLB concurrent iodine spike	0.20	0.59	3.77
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
(3) RG 1.183, Table 6
(4) 10CFR50.67
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.4-1 Steam Generator Tube Rupture (SGTR) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3659 MW_{th} (include uncertainty)
Initial RCS Equilibrium Activity (1.0μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 1.7.3-1
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate (the selected split between SGs maximizes dose)	Faulted SG – 313.33 gallons/day Intact SGs – 1126.67gallons/day
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	48 hours
RCS Mass	539,037 lb_m
SG Secondary Side Mass	99,304 lb_m per SG (minimum mass used to maximize concentration)
Release Rates	Table 2.4-2
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG (flashed tube flow) – none Faulted SG (non-flashed tube flow) – 100 Intact SGs – 100
Break Flow Flash Fraction	Table 2.4-2
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of automatic control room normal intake isolation and switch to emergency mode	Table 1.6.3-1 30 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.4-2 SGTR Release Information

Tube Break Flow - ASDV Failure Case 1

Time (hours)	Break Flow (lb_m/sec)
0.000000	12.5
0.002778	46.2
0.274167	34.9
0.500000	36.7
0.753611	42.7
1.000000	43.8
1.253611	41.4
1.461944	37.2
1.712778	37.3
1.762778	34.1
1.778333	26.2
1.793611	3.9
1.825278	4.6
1.901944	12.7
2.000000	12.6
2.777778	0

Tube Break Flow Flashing Fraction - ASDV Failure Case 1

Time (hours)	Flashing Fraction
0.000000	0.17688
0.002778	0.17864
0.274167	0.07193
0.500000	0.06080
0.753611	0.12432
1.000000	0.11501
1.253611	0.03959
1.461944	0.00229
1.712778	0.00000

Tube Break Flow - ASDV Failure Case 2

Time (hours)	Break Flow (lb_m/sec)
0.000000	12.5
0.002778	46.2
0.274167	38.1
0.416667	43.1
0.555556	43.3
0.694444	43.8
0.825000	40.1
1.032500	36.6
1.197222	37.5
1.381389	39.0
1.431389	17.3
1.495833	8.3
1.777778	2.9
1.888889	1.5
2.000000	0.0

Tube Break Flow Flashing Fraction - ASDV Failure Case 2

Time (hours)	Flashing Fraction
0.000000	0.17688
0.002778	0.17864
0.274167	0.11334
0.416667	0.14772
0.555556	0.13800
0.694444	0.12804
0.825000	0.13375
1.032500	0.05482
1.197222	0.01417
1.381389	0.00000

Intact Steam Generator Steam Release - ASDV Failure Case 1

Time (hours)	Steam Release from Unaffected SGs (lb_m/min)
0.000000	217,542
0.002778	216,967
0.274167	3,630
1.461944	9,959
1.778333	1,934
2.0	3,056
8.0	0.0
720.0	0.0

Intact Steam Generator Steam Release - ASDV Failure Case 2

Time (hours)	Steam Release from Unaffected SGs (lb_m/min)
0.0	217,542
0.002778	216,967
0.274167	4,752
0.825000	2,361
1.032500	15,738
1.197222	4,393
1.888889	4,772
2.0	3,056
8.0	0.0
720.0	0.0

Faulted Steam Generator Steam Release - ASDV Failure Case 1

Time (hours)	Steam Release from Faulted SG (lb_m/min)
0.000000	72,393
0.002778	73,140
0.274167	2,743
0.500000	11,860
0.753611	7,032
1.000000	4,843
1.253611	13.9
1.461944	0
2.0	42.6
8.0	0.0
720.0	0.0

Faulted Steam Generator Steam Release - ASDV Failure Case 2

Time (hours)	Steam Release from Faulted SG (lb_m/min)
0.000000	72,393
0.002778	73,140
0.274167	7,782
0.416667	6,446
0.555556	5,547
0.694444	4,819
0.825000	0.00
2.0	42.6
8.0	0.0
720.0	0.0

Table 2.4-3 60 μCi/gm D.E. I-131 Activities

Isotope	Activity (μCi/gm)
Iodine-131	46.36
Iodine-132	16.88
Iodine-133	74.18
Iodine-134	10.76
Iodine-135	40.80

Table 2.4-4 Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow *	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus uncertainty

Table 2.4-5 Concurrent Iodine Spike (335 x) Activity Appearance Rate

Isotope	Appearance Rate (Ci/min)
Iodine-131	140.04966
Iodine-132	158.092468
Iodine-133	271.039377
Iodine-134	212.941892
Iodine-135	211.31964

Table 2.4-6 SGTR Dose Consequences

ASDV Fails Open at 33% Level in Faulted Steam Generator (Case 1)

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)*
SGTR pre-accident iodine spike	3.78	1.85	2.07
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	2.21	1.09	1.38
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

ASDV Fails Open 3 Minutes Following Reactor Trip (Case 2)

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)*
SGTR pre-accident iodine spike	3.78	1.85	2.05
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	2.03	1.00	1.28
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
(3) RG 1.183, Table 6
(4) 10CFR50.67
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.5-1 Reactor Coolant Pump Shaft Seizure (Locked Rotor) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Core Fission Product Inventory	Table 1.7.4-1
RCS Equilibrium Activity	Table 1.7.2-1
Release Fraction from Breached Fuel	RG 1.183, Section 3.2, Table 3
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0w/o
Maximum Radial Peaking Factor	1.65
Fuel Failure	10.0%
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for fuel failure dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
Primary-to-Secondary Leakage Rate	1.0 gpm total (500 gpd maximum to any one SG)
Time to establish shutdown cooling and terminate release	8 hours
SG Minimum Mass (per SG)	99,304 lb_m
Secondary Side Iodine Activity prior to accident	Table 1.7.3-1
Secondary Side Mass Releases to environment	Table 2.5-2
Steam Generator Secondary Side Partition Coefficient	100
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of automatic control room isolation	Table 1.6.3-1 30 seconds
Breathing Rates Offsite Control Room	 RG 1.183 Section 4.1.3 RG 1.183 Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.5-2 Locked Rotor Steam Release Rate

Time (hours)	Intact SG Steam Release Rate (lb_m/min)
0.0 – 2.0	3392
2.0 – 8.0	2675
8.0 – 720.0	0

Table 2.5-3 Locked Rotor Dose Consequences

Case	EAB Dose [(1)] (rem TEDE)	LPZ Dose [(2)] (rem TEDE)	Control Room Dose [(2)] (rem TEDE)*
Locked Rotor	0.56	0.55	2.16
Acceptance Criteria	2.5	2.5	5

[(1)] Worst 2-hour dose
[(2)] Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.6-1 Rod Cluster Control Assembly (RCCA) Ejection – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0w/o
Maximum Radial Peaking Factor	1.65
% DNB Fuel	15%
% Fuel Centerline Melt	0.375%
LOCA Containment Leakage Source Term	Table 1.7.4-1
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 1.7.3-1
Release From DNB Fuel	Section 1 of Appendix H to RG 1.183
Release From Fuel Centerline Melt Fuel	Section 1 of Appendix H to RG 1.183
Steam Generator Secondary Side Partition Coefficient	100
Steam Generator Tube Leakage	1.0 gpm total
Time to establish shutdown cooling and terminate steam release	8 hours
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for fuel failure dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
SG Secondary Side Mass	minimum – 99,304 lb_m (one SG) Minimum mass used for SGs to maximize steam release nuclide concentration.
Chemical Form of Iodine Released to Containment	Particulate – 95% Elemental – 4.85% Organic – 0.15%
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97 % Organic – 3%
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Time of Control Room Ventilation System Isolation	30 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Input/Assumption	Value
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.704E+06 ft^3 0.15% (by weight)/day 0.075% (by weight)/day
Secondary Containment (CEVA) Filter Efficiency	Particulate – 95% Elemental – 95% Organic – 85%
Secondary Containment (CEVA) Drawdown Time	360 seconds
Secondary Containment (CEVA) Bypass Fraction	60%
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 2.2 hr^{-1} Organic Iodine – None

Table 2.6-2 RCCA Ejection Steam Release Rate

Time (hours)	SG Steam Release Rate (lb_m/min)
0.0 – 2.0	3392
2.0 – 8.0	2675
8.0 – 720.0	0

Table 2.6-3 RCCA Ejection Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)*
RCCA Ejection – Containment Release (190 cfm CR unfiltered inleakage)	1.69	1.95	4.92
RCCA Ejection – Secondary Release (300 cfm CR unfiltered inleakage)	1.52	1.44	3.86
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 190 cfm for the containment release case and 300 cfm for the secondary release case
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.7-1 Letdown Line Rupture – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 1.7.3-1
Iodine spike appearance rate	500 times (See Table 2.3-5 for values)
Duration of accident initiated spike	8 hrs
Condenser Decontamination Factor	100
Steam Generator Tube Leakage	1 gpm (total for all SGs)
RCS mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for iodine spike dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
Letdown Line Rupture flow rate	140 gpm (1160 lb/min) for 30 minutes
Letdown Line Flashing Fraction	0.1815 at 380 F and 2235 psia
Letdown Line Rupture Release Point	Worst release point (to the CR) from the PAB
Secondary Side Release Point	Worst release point (to the CR) from the Turbine Building (condenser)
Control Room Ventilation System Time of automatic control room isolation	Table 1.6.3-1 30 seconds
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Breathing rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

Table 2.7-2 Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow *	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus 10% uncertainty

Table 2.7-3 Letdown Line Rupture Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)*
Letdown Line Rupture	0.46	0.27	1.54
Acceptance Criteria	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
(3) RG 1.183, Table 6
(4) 10CFR50.67
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.8-1 Radioactive Gaseous Waste System Failure – Inputs and Assumptions

Input/Assumption	Value
RGWS release inventory	Table 2.9-2
RGWS component volume (arbitrary)	10,000 ft^3
Tank leak rate (arbitrarily high)	Entire inventory released within 2 hours
Control Room Ventilation System Time of automatic control room isolation	Table 1.6.3-1 30 seconds
Atmospheric Dispersion Factors Offsite Onsite	Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Breathing Rates Offsite Control Room	RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

Table 2.8-2 RGWS Source Term *

Isotope	RGWS Inventory (Curies)
Kr-83m	8.2E+01
Kr-85m	8.1E+02
Kr-85	9.0E+02
Kr-87	1.5E+02
Kr-88	1.0E+03
Xe131m	2.2E+03
Xe-133m	3.6E+03
Xe-133	3.8E+05
Xe-135m	1.2E+01
Xe-135	3.2E+03
Xe-137**	2.1E-01
Xe-138	8.3E+00

* from UFSAR Table 15.7-3
** not included in analysis (insignificant)

Table 2.8-3 RGWS Failure Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)*
RGWS	0.89	0.43	1.02
Acceptance Criteria	2.5	2.5	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 2.9-1 Radioactive Liquid Waste System Failure – Inputs and Assumptions

Input/Assumption	Value
RLWS release inventory	Table 2.10-2 and 2.10-3
RLWS component volume (arbitrary)	10,000 ft^3
RWLS leak rate (arbitrarily high)	Entire inventory released within 2 hours
Control Room Ventilation System Time of automatic control room isolation	Table 1.6.3-1 30 seconds
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

Table 2.9-2 RLWS Source Term *

Isotope	Letdown Degasifier Release (Curies)	Boron Waste Tank Release (Curies)
I-131	1.7E-01	7.8E-00
I-132	5.5E-02	1.0E-00
I-133	2.2E-01	8.8E-00
I-134	3.0E-02	3.3E-01
I-135	1.4E-01	4.0E-00
Kr-83m	2.6E+01	--
Kr-85m	1.1E+02	--
Kr-85	9.2E+00	--
Kr-87	7.3E+01	--
Kr-88	2.2E+02	--
Xe-131m	4.7E+00	1.3E-01
Xe-133m	4.0E+01	1.5E-01
Xe-133	1.8E+03	4.9E+00
Xe-135m	2.6E+01	1.2E+01
Xe-135	2.1E+02	7.5E+00
Xe-137**	2.0E+00	--
Xe-138	2.1E+01	--

* from UFSAR Table 15.7-10
** not included in analysis (insignificant)

Table 2.9-3 RLWS Failure Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)*
Boron Waste Tank	0.04	0.02	0.92
Letdown Degasifier	0.04	0.02	0.46
Acceptance Criteria	2.5	2.5	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
* Control Room dose includes conservative radiation shine contribution of 0.45 rem

Table 3-1

Seabrook Station
Summary of Alternative Source Term Analysis Results

Case	Allowable Unfiltered CR Inleakage (cfm)	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	150	4.40	3.38	4.73
MSLB - Pre-Accident Iodine Spike	300	0.07	0.11	1.11
SGTR - Pre-accident Iodine Spike (Case 1 is limiting)	300	3.78	1.85	2.07
Acceptance Criteria		≤ 25(3)	≤ 25(3)	≤ 5(4)
MSLB - Concurrent Iodine Spike	300	0.20	0.59	3.77
SGTR - Concurrent Iodine Spike (Case 1 is limiting)	300	2.21	1.09	1.38
Locked Rotor	300	0.56	0.55	2.16
Small Line Break Outside Containment (Letdown Line) *	300	0.46	0.27	1.54
Radioactive Gaseous Waste System Failure *	300	0.89	0.43	1.02
Radioactive Liquid Waste System Failure * (Boron Waste Tank is limiting)	300	0.04	0.02	0.92
Acceptance Criteria		≤ 2.5 (3)	≤ 2.5 (3)	≤ 5 (4)
FHA	300	1.41	0.69	2.39
RCCA Ejection – Containment Release	190	1.69	1.95	4.92
RCCA Ejection – Secondary Side Release	300	1.52	1.44	3.86
Acceptance Criteria		≤ 6.3(3)	≤ 6.3(3)	≤ 5(4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) RG 1.183, Table 6
(4) 10CFR50.67

* see Section 1.4 and appropriate event summary in Section 2.0 for basis of acceptance criteria