Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

September 18, 2003

L-2003-224
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendments
Alternate Source Term and Conforming Amendments

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License DPR-67 for St. Lucie Unit 1. FPL proposes to revise the St. Lucie Unit 1 licensing bases to adopt the alternate source term (AST) as allowed in 10 CFR 50.67. This application also includes conforming amendments to the Unit 1 Technical Specifications 1.10, 3.4.6.2, and 6.8.4.h.

Attachment 1 is a description of the proposed changes and the supporting justification. Attachment 2 is the Determination of No Significant Hazards and Environmental Considerations. Attachment 3 is marked up copies of the proposed Technical Specification changes. Attachment 4 is copies of the retyped TS pages. Enclosure 1 is Numerical Applications, Inc. Report NAI-1101-043, *AST Licensing Technical Report for St. Lucie Unit 1*, Revision 2.

The St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board have reviewed the proposed amendment. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

Approval of this proposed license amendment is requested within one year of submittal to support FPL planned actions to resolve Generic Letter 2003-01, *Control Room Habitability*. Please issue the amendment to be effective on the date of issuance and to be implemented within 60 days of receipt by FPL. Please contact George Madden at 772-467-7155 if there are any questions about this submittal.

Very truly yours,

William Jefferson, Jr.
Vice President
St. Lucie Plant

WJ/GRM

Attachments

A001

cc: Mr. William A. Passetti, Florida Department of Health

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

William Jefferson, Jr. being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

William Jefferson, Jr.

STATE OF FLORIDA

COUNTY OF ST LUCIE

Sworn to and subscribed before me

this 18 day of Sept, 2003
by William Jefferson, Jr., who is personally known to me.

Leslie J. Whitwell
Name of Notary Public - State of Florida

Leslie J. Whitwell
MY COMMISSION # DD020212 EXPIRES
May 12, 2005
BONDED THRU TROY FAIN INSURANCE, INC.

(Print, type or stamp Commissioned Name of Notary Public)

ATTACHMENT 1

Regulatory Assessment of the Proposed Implementation of the Alternative Radiological Source Term Methodology For The St. Lucie Unit 1.

Introduction

The current St. Lucie Unit 1 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.

Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10 CFR 50.67 was issued to allow holders of operating licenses to voluntarily revise the traditional accident source term used in the design basis accident radiological consequence analyses with alternative source term (AST). 10 CFR 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 AST is provided in Regulatory Guide (RG) 1.183.

As documented in Nuclear Energy Institute (NEI) guidance document, NEI 99-03, several nuclear plants performed testing on control room unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in the accident analyses. The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for St. Lucie Unit 1 to support the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage.

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)
- Control Element Assembly (CEA) Ejection
- Inadvertent Opening of a Main Steam Safety Valve (IOMSSV)
- Spent Fuel Cask Drop, and
- Waste Gas Decay Tank (WGDT) Rupture.

Each accident and the specific input assumptions are described in the Numerical Applications, Inc. (NAI) Report NAI-1101-043, *AST Licensing Technical Report for St. Lucie Unit 1*, Revision 2. These analyses provide for a bounding allowable control room unfiltered air inleakage of 580 cfm. The use of 580 cfm as a design basis value is

expected to be well above the unfiltered inleakage value to be determined through testing or analysis consistent with the resolution of issues identified in NEI 99-03.

Description of Proposed Amendment

Florida Power & Light Company proposes to revise the St. Lucie Unit 1 licensing basis to implement the AST, described in RG 1.183, 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 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- 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).
- A steam generator tube leakage rate that is more restrictive than the current Technical Specification limit is utilized.
- A shield building ventilation system (SBVS) bypass leakage value that is more restrictive than the current Technical Specification limit is utilized.

Accordingly, the following changes to the St. Lucie, Unit 1 Technical Specifications (TS) are 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.

- The reactor coolant system (RCS) operational leakage limits stated in Limiting Condition for Operation (LCO) 3.4.6.2 for total primary-to-secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm and 216 gallons per day (gpd) through any one generator.

- The leakage rate acceptance criterion for secondary containment bypass leakage paths (i.e. Shield Building Bypass Leakage) stated in TS 6.8.4.h, Containment Leakage Rate Testing Program, is reduced from 27% to 9.6% as supported by plant testing results.

Accident Source Term

The full core isotopic inventory for St. Lucie Unit 1 is determined in accordance with RG 1.183. The inventory of fission products in the core and coolant systems that is available

for release to the containment is based on the maximum full power operation of the core and the current licensed values for fuel enrichment, and fuel burnup. Event-specific isotopic source terms are developed using a bounding approach. The maximum core power of 2754 MW_{th} is calculated as the current licensed rated thermal power of 2700 MW_{th} plus the emergency core cooling system (ECCS) evaluation uncertainty of 2%. 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, 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 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 TS definition of Dose Equivalent I-131 (DE I-131) and dose conversion factors for individual isotopes from ICRP 30 (which are equivalent to the rounded values from FGR No. 11 for iodine isotopes). The remaining (non-iodine) isotopes are adjusted to achieve the TS 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 TS. 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 1/10 of the primary coolant activity.

The fuel handling accident for St. Lucie Unit 1 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 102% of rated power (2754 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 St. Lucie Unit 1 (NAI-1101-043) provides the details of the LOCA and non-LOCA accident analyses performed according to the guidelines set forth in RG 1.183.

Dose Calculation

The St. Lucie Unit 1 dose calculations using the AST methodology apply TEDE acceptance criteria. Dose calculations follow the guidelines of Regulatory Positions cited in RG 1.183.

Analyses consider the radionuclides listed in Table 5 of RG 1.183 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 iodine, 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 St, Lucie Unit 1 use assumptions and models defined in RG 1.183 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. In some cases, St. Lucie Unit 1 has opted to not take credit for a qualified accident mitigation feature in order to provide an additional measure of conservatism. NAI-1101-043 describes these exceptions. Selected numeric input values are conservative to assure a conservative calculated dose. Except as otherwise required by regulatory guidance, analyses use current licensing basis values.

Meteorological data collected by the St. Lucie Unit 1 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, into the design basis accident analysis is made to support control room habitability with increased control room unfiltered air inleakage. Analysis of the dose consequences of the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, IOMSSV, Spent Fuel Cask Drop, and WGDT Rupture are made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the St. Lucie Unit 1 licensing basis and the calculated doses do not exceed the defined acceptance criteria.

Results of the St. Lucie Unit 1 radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered inleakage are summarized in Table 1. The analyses support a maximum allowable control room unfiltered air inleakage of 580 cfm. The enclosed NAI-1101-043, *AST Licensing Technical Report for St. Lucie Unit 1*, explains these results and acceptance criteria in more detail.

References

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

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

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

4. NAI-1101-043, *AST Licensing Technical Report for St. Lucie Unit 1*, Revision 2, Numerical Applications, Inc., August 20 2003.

5. 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.

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

Table 1
St. Lucie Unit 1
Summary of Alternative Source Term (AST) 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	580	1.20	2.56	4.88
MSLB – Outside of Containment (3.5% DNB)	580	0.31	0.85	4.87
MSLB – Outside of Containment (0.81% FCM)	580	0.35	0.89	4.86
MSLB – Inside of Containment (100% DNB)	580	0.32	0.80	3.86
MSLB – Inside of Containment (16% FCM)	580	0.59	1.29	4.75
SGTR Pre-Accident Iodine Spike	1000	0.29	0.28	3.31
Acceptance Criteria		$\leq$ 25[3]	$\leq$ 25[3]	$\leq$ 5[4]
SGTR Concurrent Iodine Spike	1000	0.10	0.10	1.10
Locked Rotor (13.7% DNB)	1000	0.10	0.23	1.64
IOMSSV *	1000	0.02	0.02	0.40
Acceptance Criteria		$\leq$ 2.5[3]	$\leq$ 2.5[3]	$\leq$ 5[4]
FHA	1000	0.33	0.32	3.72
CEA Ejection – Containment Release (9.5% DNB, 0.5% FCM)	1000	0.13	0.28	1.67
CEA Ejection – Secondary Side Release (9.5% DNB, 0.5% FCM)	1000	0.23	0.47	3.03
Spent Fuel Cask Drop – Case 1*	1000	4.5370E-04	0.20	2.42
Spent Fuel Cask Drop – Case 2*	1000	3.3549E-04	0.20	2.39
WGDT	1000	0.17	0.17	0.50
Acceptance Criteria		$\leq$ 6.3[3]	$\leq$ 6.3[3]	$\leq$ 5[4]

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) RG 1.183, Table 6
(4) 10 CFR 50.67

* see appropriate event summary in NAI-1101-043 for bases of acceptance criteria

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Proposed Change

Florida Power & Light Company (FPL) proposes to revise the St. Lucie Unit 1 licensing basis to implement the alternate source term (AST), described in Regulatory Guide (RG) 1.183, through reanalysis of the radiological consequences of the following limiting Updated Final Safety Analysis Report (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)
- Control Element Assembly (CEA) Ejection
- Inadvertent Opening of a Main Steam Safety Valve (IOMSSV)
- Spent Fuel Cask Drop, and
- Waste Gas Decay Tank (WGDT) Rupture.

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 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- 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 is being used to calculate the offsite and control room radiological consequences for St. Lucie Unit 1 to support the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage.
- A steam generator tube leakage rate that is more restrictive than the current Technical Specification (TS) limit is utilized.
- An shield building ventilation system (SBVS) bypass leakage value that is more restrictive than the current Technical Specification limit is utilized.

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

- The definition of Dose Equivalent I-131 in Section 1.10 is revised to reference Federal Guidance Report 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.

- The Reactor Coolant System (RCS) operational leakage limits, stated in Limiting Condition for Operation (LCO) 3.4.6.2 for total primary-to-secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm, not to exceed 216 gallons per day (gpd) per steam generator.

- The leakage rate acceptance criterion for secondary containment bypass leakage paths (i.e. Shield Building Bypass Leakage) stated in TS 6.8.4.h, Containment Leakage Rate Testing Program, is reduced from 27% to 9.6% as supported by plant testing results.

Determination of No Significant Hazards Consideration

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

This change does not involve a significant hazards consideration for the following reasons:

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

   Alternative source term calculations have been performed for St. Lucie Unit 1 that demonstrate the dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed change does not modify the design or operation of the plant. The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of any accident. The AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed changes to the Technical Specifications, demonstrate that the dose consequences of these limiting events are all within the regulatory limits. The proposed Technical Specification changes to the RCS operational leakage limits and to the shield

building bypass leakage rate acceptance criterion result in more restrictive requirements and support the AST revisions to the limiting design basis accidents.

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

2. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident 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 more restrictive proposed leakage limits 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 amendment does not involve a significant reduction in the margin of safety.**

   The proposed implementation of the alternative source term methodology is consistent with NRC Regulatory Guide 1.183. The Technical Specification changes to the RCS operational leakage limits and to the shield building bypass leakage rate acceptance criterion result in more restrictive requirements and support revisions to the radiological analyses of the limiting design basis accidents. Conservative methodologies, per the guidance of RG 1.183, 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 and 10 CFR 50.67. 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 10 CFR 50.67 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 discussion, FPL has determined that the proposed change does not involve a significant hazards consideration.

Environmental Consideration

10 CFR 51.22(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. FPL has reviewed this proposed license amendment request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(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. The basis for this determination follows:

Proposed Change

St. Lucie Unit 1 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.

Basis

This change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the 10 CFR 50.92 evaluation, 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 purely 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.

ATTACHMENT 3

ST. LUCIE UNIT 1 MARKED-UP TECHNICAL SPECIFICATION PAGES

TS Pages

1-3

3/4 4-14

6-15b

DEFINITIONS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (µCi/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 ~~ICRP-30, Supplement to Part 1, pages 192-212, Tables entitled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity (Sv/Bq)."~~ 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

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES REPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF 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.

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 0.3gpm ~~1 GPM~~ total primary-to-secondary leakage through steam generators, and 216 ~~gpd~~ gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," as modified by the following exception(s):

a) Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) will be used for type A testing.

b) The first Type A test performed after the May 1993 Type A test shall be no later than May 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 39.6 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

BLIND NOTE PLA L-2003-199
Corrects typo to 0.60

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < ~~6.0~~ L_a for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq$ ~~0.27~~ L_a for secondary containment bypass leakage paths.

0.096

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For the personnel air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq 1.0\ P_a$.

3) For the emergency air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to ≥ 10 psig.

ATTACHMENT 4

ST. LUCIE UNIT 1 RETYPED TECHNICAL SPECIFICATION PAGES

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.

TS Pages

1-3

3/4 4-14

6-15b - (Also changed by PLA L-2003-199)

DEFINITIONS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES REPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF 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.

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 0.3 gpm total primary-to-secondary leakage through steam generators and 216 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) *Participation in a Interlaboratory Comparison Program* to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," as modified by the following exception(s):

a) Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) will be used for type A testing.

b) The first Type A test performed after the May 1993 Type A test shall be no later than May 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 39.6 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 6.0\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.096\ L_a$ for secondary containment bypass leakage paths. |

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For the personnel air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq 1.0\ P_a$.

3) For the emergency air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to ≥ 10 psig.

Enclosure 1

Numerical Applications, Inc.

NAI Report No. NAI-1101-043

AST Licensing Technical Report for St. Lucie Unit 1

Revision 2

Dated August 20, 2003

NAI Report Release

Report Number: NAI-1101-043

Revision Number: 2

Title: AST Licensing Technical Report for St. Lucie Unit 1

Description:

This report documents the results of the analyses and evaluations performed by Numerical Applications, Inc. in support of the St. Lucie Unit 1 licensing 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 (Merv Marshall)	8/19/03 Date
Reviewer (Steve Thomasson)	8/19/03 Date
Reviewer (Joe Sinodis)	8/19/03 Date
NAI Management (Jim Harrell)	8/20/03 Date

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Figures and Tables

1.0 Radiological Consequences Utilizing the Alternative Source Term Methodology

1.1 Introduction

The current St. Lucie Plant, Unit No. 1, 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 is issued to allow 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).

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 the 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 St. Lucie Unit No. 1 to support the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage.

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)
- Control Element Assembly (CEA) Ejection
- Inadvertent Opening of a Main Steam Safety Valve (IOMSSV)
- Spent Fuel Cask Drop, and
- Waste Gas Decay Tank (WGDT) Rupture*.

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 580 cfm. The use of 580 cfm as a design basis value is expected to be well above the unfiltered inleakage value to be determined through testing or analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01.

* Although the consequences of the WGDT Rupture event are based on Technical Specification activity limits and are not dependent upon reactor power or the reactor core source term, the analysis is updated to incorporate new atmospheric dispersion factors and to evaluate the dose consequences to the revised TEDE criteria.

1.3 Proposed Changes to the St. Lucie Unit No. 1 Licensing Basis

Florida Power and Light (FP&L) Company proposes to revise the St. Lucie Plant, Unit No. 1, 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 1.1 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).
- A steam generator tube leakage rate that is more restrictive than the current Technical Specification limit is utilized.
- An SBVS bypass leakage value that is more restrictive than the current Technical Specification limit is utilized. Plant maintenance and surveillance history demonstrate that the proposed reduced containment leakage values have been met in the past (Reference 5.8).

Accordingly, the following changes to the St. Lucie, Unit No. 1, Technical Specifications (TS) are 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. 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).

- The Reactor Coolant System (RCS) operational leakage limits, stated in Limiting Condition for Operation (LCO) 3.4.6.2 for total primary-to-secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm, not to exceed 0.15 gpm per steam generator.

- The leakage rate acceptance criterion for secondary containment bypass leakage paths (i.e. Shield Building Bypass Leakage) stated in TS 6.8.4.h, "Containment Leakage Rate Testing Program," is reduced from 27% to 9.6% as supported by plant testing results.

1.4 Compliance with Regulatory Guidelines

The revised St. Lucie Unit No. 1 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 WGDT Rupture dose consequences acceptance criteria for EAB and LPZ based on FHA acceptance criteria (6.3 rem). Precedent is established in the Kewaunee Nuclear Power Plant AST submittal dated March 19, 2002 and subsequent Issuance of Amendment (IA) and Safety Evaluation (SE) issued March 17, 2003. This is also consistent with the current licensing basis, where WGDT and FHA are both Class 3 events.

- 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.

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	5.11	Atmospheric Dispersion Factors
MicroShield	5.05	5.12	Direct Shine Dose Calculations
ORIGEN	2.1	5.13	Core Fission Product Inventory
PAVAN	2.0	5.14	Atmospheric Dispersion Factors
RADTRAD-NAI	1.0p3	5.15	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.

1.5.2 MicroShield – used to analyze shielding and estimate exposure from gamma radiation.

1.5.3 ORIGEN – used for calculating the buildup, decay, and processing of radioactive materials.

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.

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).

RADTRAD-NAI is 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 St. Lucie Unit No. 1, the events not specifically addressed in RG 1.183 are the Inadvertent Opening of a MSSV, the Spent Fuel Cask Drop, and the Waste Gas Decay Tank (WGDT) Rupture.

1.6.3 Control Room Ventilation System Description

The Control Room Air Conditioning System (CRACS) and Control Room Emergency Ventilation System (CREVS) are required to assure control room habitability. The design of the control room envelope and overall descriptions of the Control Room Air Conditioning and Emergency Ventilation Systems are discussed in the St. Lucie Unit 1 UFSAR, Sections 6.4 and 9.4.1, and a system airflow diagram is shown on UFSAR Figure 9.4-1.

The control room envelope is pressurized relative to the surroundings at all times during normal plant operation with outside air continuously introduced to the control room envelope at a rate of 750 cfm (for dose analyses purposes, this value is conservatively increased to 920 cfm). Following a design basis accident the control room is pressurized at the rate of 450 cfm to maintain a positive pressure differential. Makeup air for pressurization is filtered before entering the control room.

Automatically actuated, redundant isolation valves are provided at each outside air intake and exhaust air path so that the control room envelope is immediately isolated on receipt of a CIAS, or outside air intake high radiation signal. Unfiltered inleakage paths through the isolation valves are reduced by using low leakage butterfly valves. The Control Room Air Conditioning System is capable of automatic actuation or manual transfer from its normal operating mode to the pressurized or isolated modes as necessary. The system is designed to perform its safety functions and maintain a habitable environment in the control room envelope during isolation.

The net volume of the control room envelope serviced by the Control Room Air Conditioning and Emergency Ventilation Systems is approximately 62,700 cubic ft.

The Control Room Air Conditioning System (CRACS) consists of three air conditioners and a ducted air intake and air distribution system. The system is zone isolated with filtered recirculated air, widely separated dual air inlets, and provisions for positive pressurization (≥1/8 in. water gauge). Each air conditioner includes a cabinet type centrifugal fan, direct expansion refrigerant cooling coil, roughing filter, refrigerant condenser and refrigerant compressor. Air conditioning unit capacity is 50 percent

each during normal operation and 100 percent each during post LOCA operation. Under emergency conditions, only one out of three air conditioning units and one train of the Control Room Emergency Ventilation System are required to maintain the habitability of the control room envelope.

The habitability systems (air filtration and ventilation equipment with associated instrumentation, controls and radiation monitoring) are capable of performing their functions assuming a single active component failure coincident with a loss of offsite power. Redundant equipment which is essential to safety is powered from separate safety related buses such that loss of one bus does not prevent the Control Room Air Conditioning System from fulfilling its safety function.

The control room operator has the ability, through radiation monitors, to determine radiation levels in each of the outside air intake ducts. Radiation monitors are located at both outside air intakes. Additional radiation monitoring capability is provided in the control room.

The Control Room Emergency Ventilation System (CREVS) consists of prefilters, High Efficiency Particulate (HEPA) prefilters, charcoal adsorbers, HEPA after filters and centrifugal fans.

1.6.3.1 Normal Operation

During normal operation, the control room is air conditioned by the air conditioning units. Two of the three air conditioning units in the control room are in the automatic mode of control. One or two units are normally running with the other unit(s) in a standby status, available for manual actuation in the event of a failure of an operating unit. Fresh air is taken in through either the northern or the southern outside air intakes by remote manual opening of the redundant motor operated isolation valves.

Control room air is drawn into the air handling section through a return air duct system and roughing filters (not credited for dose analyses), and is cooled as required. Conditioned air is directed back to the control room through the supply air duct system. Outside air makeup is effected through either of two outside air intakes located in the northern and southern walls of the Reactor Auxiliary Building at elevation 78 feet 9 inches. This makeup air replenishes the air exfiltrated to the outside in addition to that being exhausted by the toilet and kitchen exhaust fans. The return air flowrate is controlled by dampers to maintain a constant positive pressure of 1/8 inch wg in response to the average pressure differential between the control room and its surroundings.

During normal operation the CREVS is isolated from the CRACS ducts by dampers.

1.6.3.2 Emergency Operation

The emergency modes of operation of the Control Room Air Conditioning System are:

a) automatic isolation and automatic recirculation with filtration of recirculated air, or

b) automatic isolation with immediate manual and/or automatic filtered pressurization and recirculation with filtration.

Upon receipt of a containment isolation actuation signal (CIAS) or a high radiation signal, the redundant isolation valves on the outside air intake and exhaust ducts close automatically. The two CREVS filtration units start automatically while the air conditioning units remain running. The isolation time including the damper closure time is equal to a maximum of 35 seconds, assuming offsite power is available. If offsite power is not available, the isolation time is 45 seconds, which includes the 10-second diesel generator start time (for dose analyses purposes, this value is conservatively increased to 50 seconds). A portion of the control room air is recirculated through the HEPA filters and charcoal adsorbers for removal of radioactive particles and iodine, respectively.

Outside air intake dampers are adjusted to allow sufficient outside air makeup flow to maintain control room pressurization. By observing the radiation monitors located in the outside air intake ducts, the operator restores outside air makeup by selecting which set of isolation valves to open. After determining which outside air intake has the least, or zero, amount of radiation, the operator opens the isolation valves on that intake and adjusts the system dampers for proper flow. All outside air make-up and a portion of the control room return air is passed through a filter train for removal of radioactive particulates, iodides, carbon dioxide and other gaseous impurities before it enters the air conditioning units. Depending on the cooling required the operator may stop or start air conditioning units. The operator stops one of the two CREVS filtration units.

In the event of a CIAS or high radiation signal followed by a loss of offsite power, the outside air intake isolation valves are designed to fail as is and the CREVS fans stop. Outside air is not drawn into the control room because the control room is pressurized during normal operation and the coasting down fan is discharging against a positive pressure in addition to overcoming ductwork and damper frictional losses. When sequenced onto the diesel generator the valves automatically close and the CREVS fan is started.

The Control Room Emergency Ventilation System removes potentially radioactive particulates and iodine from the control room air during the post-LOCA operating mode. The unit consists of a roughing filter, HEPA prefilter, charcoal adsorber, HEPA after-filter and fans. The system operates post-LOCA to maintain a positive control room pressure. The flow control valves, installed in each air intake, control the flow of air being drawn into the control room. Post-LOCA makeup flow enters through one of these ducts and passes through the charcoal filters. Thus, all makeup air is filtered.

The air cleaning unit removes radioactivity from the control room envelope atmosphere. The HEPA filters remove 0.3-micron particles from atmospheric air at an efficiency greater than 99.9 percent. The charcoal adsorbers have an elemental and organic iodine removal efficiency of 99.825 percent minimum.

1.6.3.3 Control Room Dose Calculation Model

The Control Room model includes a recirculation filter model along with filtered air intake, unfiltered air inleakage 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 St. Lucie Unit 1 plant 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 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 either or both of two ventilation intake locations that are located on opposite ends of the CR. 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 is described in subsequent event analyses sections.

During normal operation, both of these control room ventilation intakes are open and the control room ventilation draws in 750 cfm of fresh outside air through both of the vents in parallel and delivers it unfiltered to the control room. For AST analyses this value is being conservatively increased to 920 cfm. In this configuration, the dispersion factor for air being drawn into the control room is assumed to be the average of the dispersion factors for the two intake locations. These intakes are both automatically closed upon actuation of the CR isolation mode and no air is intentionally drawn into the control room ventilation. However, the control room ventilation system recirculates the air within the CR through the filtration system to remove contaminates that are already drawn into the system or have leaked into the control room. During the course of the event, fresh air is required to be added to the CR in order to maintain positive pressure and air quality. The operator will selectively open the ventilation system intake location with the lower radioactive concentrations and draw 450 cfm of outside air through the filtration system and into the control room. Therefore, at this point, the model uses the dispersion factor for the more favorable air intake location for assessing the dose from this filtered makeup contribution. This filtered intake is assumed to continue throughout the rest of the 30-day duration of the dose calculation.

During the entire course of the event it is also assumed that contaminated outside air can also enter the control room (unfiltered) via various leakage paths. This air may enter the control room through a number of different locations that may be defined by testing. In the absence of detailed testing results, some judgements are necessary in order to assign a single dispersion factor that is appropriate for the combined unfiltered inleakage from various diverse sources. At the beginning of the event, the dose calculation conservatively assigns an initial dispersion factor applicable to the least favorable control room ventilation system intake location. Following CR isolation, when both CR ventilation intakes are closed, the dispersion factor for the CR unfiltered inleakage assumes a dispersion factor corresponding to a location that is at the midpoint of both of the CR intake locations. At the time when the operator unisolates the control room by opening the favorable air intake, this analysis will apply the dispersion factor for the more favorable CR intake location to the unfiltered inleakage component.

The timing of the above evolutions is dependent upon the event. However, Control Room isolation is initiated by either a CIAS signal (generated by containment pressure or SI) or a high radiation signal at the CR intake. For the LBLOCA case, the containment isolation signal reached within a few seconds into the blowdown will also trigger Control Room isolation. The model imposes a 50-second delay to allow the CR ventilation system to physically switch into isolation mode. This delay is conservative with respect to the time required for signal processing, relay actuation, time required for the dampers to move and the system to re-align, and diesel generator start time.

The time at which the operator will act to unisolate the control room and initiate the 450 cfm of filtered air makeup is a proceduralized operator decision during the course of the event. For St. Lucie Unit 1 the nominal time to unisolate the CR is assumed to be 90 minutes from the start of the event based on past experience and procedures.

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 St. Lucie Unit 1 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 LBLOCA 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. Previous analysis results verifying this assumption for St. Lucie Unit 2 are listed in Table 6.4-2 of the St. Lucie Unit 2 UFSAR. It is reasonable to make the same assumption for St. Lucie Unit 1 considering the nearly identical layout of the unit sites and hypothesized accident conditions.

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 where they are either used as is, or 'decayed' (once the release has stopped) in MicroShield to yield the source activity at a later point in time. 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 St. Lucie Unit 1 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
Table 1.7.6-1 - Spent Fuel Cask Drop 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 cases for individual event analysis. For example, the RADTRAD 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 St. Lucie Unit No. 1 reactor core consists of 217 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 (part of the SCALE-4.3 system of codes) 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 102% of rated power (102% of 2700 MW_{th}, or 2754 MW_{th}). For rod average burnups in excess of 54,000 MWD/MTU the heat generation rate is limited to 6.3 kw/ft. For non-LOCA events with fuel failures, a bounding radial peaking factor of 1.7 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 (424,160 grams) is assumed to apply to all 217 fuel assemblies in the core.

2. Radioactive decay of fission products during refueling outages is ignored.

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 St. Lucie Unit 1 is derived from Table 11.1-1 of the UFSAR. Per the assumptions listed in Table 11.1-2 of the UFSAR, the activities given in Table 11.1-1 are based on 1% failed fuel. Table 11.1-1of the UFSAR presents the activities in units of µCi/cc for 70°F water. The density of 70°F water is 1.0 gm/cc; therefore, 1.0 µCi/cc is equal to 1.0 µCi/gm.

The iodine activities from UFSAR Table 11.1-1 are adjusted to achieve the Technical Specification limit of 1.0 µCi/gm dose equivalent I-131 using the Tech. Spec. definition of Dose Equivalent I-131 (DE I-131) and dose conversion factors for individual isotopes from ICRP 30 (which are equivalent to the rounded values from FGR 11 for iodine isotopes). The non-iodine species are adjusted to achieve the Technical Specification limit of 100/E-bar for non-iodine activities. The remaining isotopes are adjusted to achieve the Tech. Spec. limit of 100/E-bar microcuries per gram of gross activity.

The dose conversion factors for inhalation and submersion are from Federal Guidance Reports Nos. 11 and 12 respectively. With respect to ICRP 30, FGR 11 states: The ALI (Annual Limit on Intake) and DAC (Derived Air Concentration) values tabulated in FGR 11 are identical to those of ICRP 30, except for the isotopes of Np, Pu, Am, Cm, Bk, Cf, Es, Fm, and Md. RIS 2001-19 states that the NRC staff considers thyroid dose conversion factors based on ICRP-30, such as those tabulated in FGR 11, to be an acceptable change in methodology that does not warrant prior review. FGR 12 explains that the submersion dose coefficient data given in FGR 11 Table 2.3 are based on the dosimetric analysis of ICRP 30. FGR 12 goes on to state that the submersion coefficients of Table III.1 of FGR 12 are an improvement over those in Table 2.3 of FGR 11 (differences described on pages 55 and 56 of FGR 12), producing only minor differences in the effective dose.

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 1/10 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 (non-LOCA)."

1.7.4 LOCA Containment Leakage Source Term

Per Section 3.1 of RG 1.183, the inventory of fission products in the St. Lucie Unit 1 reactor core and available for release to the containment is based on the maximum full power operation of the core (2754 MW_{th} = 2700 + 2% uncertainty) and the current licensed 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 12.691 MW_{th}. The minimum fuel enrichment is based on an historical minimum of 3.0 w/o

and the maximum fuel enrichment is the Tech. Spec. maximum value of 4.5 w/o. It is conservatively assumed that a maximum assembly uranium mass of 424,160 gm applies to all of the fuel assemblies.

*Average assembly power = (2700 MW_{th})(1.02)(1 / 217 assemblies) = 12.691 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 St. Lucie Unit 1 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 RG 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 1.7 (total integrated peaking factor per St. Lucie Unit 1 COLR), is applied in determining the inventory of the damaged rods.

The LOCA containment leakage source term is based on the activity of 217 fuel assemblies and the radial peaking factor is 1.7. Thus, based on the methodology specified in RG 1.183, the fuel handling accident source term is derived by applying a factor of 1.7/217 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 both 3.0 w/o and 4.5 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."

1.7.6 Spent Fuel Cask Drop Source Terms

Section 9.1.4.3.d of the UFSAR describes two cask drop cases:

<u>Case 1</u>
"One-third of a core is placed in the spent fuel pool each year during refueling for the next 23 years, until the pool is filled (7 2/3 cores). The number of assemblies damaged is equal to the number offloaded during a normal refueling plus the remainder of the pool filled with discharged assemblies from the previous refuelings."

<u>Case 2</u>
"One-third of a core is placed in the spent fuel pool each year during refueling for the next 20 years. Following the 21st year of operation, the entire core is removed from the reactor and placed into the pool, which fills the pool (7 2/3 cores). The number of assemblies damaged is equal to a full-core offload plus the remainder of the pool filled with discharged assemblies from the previous refuelings."

Consistent with the requirements of Technical Specification 3.9.14, Case 1 assumes a decay time of 1180 hours for the last fuel assemblies offloaded to the pool and Case 2 assumes a decay time of 1490 hours for the last full core offloaded to the pool. The decay time of the remainder of the fuel

assemblies is assumed to be consistent with their discharge history. Decay time between cycles is conservatively ignored

The cask drop source term is based on the activity of 217 fuel assemblies. A maximum Uranium mass of 424,160 grams is assumed to apply to all discharged assemblies. To ensure that the "bounding" assembly is identified, the activity of an average burnup assembly (45,000 MWD/MTU), at both 3.0 w/o and 4.5 w/o, is determined. ORIGEN is used to determine the activities at one-year intervals for 22 years after discharge. For each nuclide, the bounding activity for the allowable range of enrichments and discharge exposure is determined.

The Spent Fuel Cask Drop source terms are presented in Table 1.7.6-1.

For Cask Drop Case 1, the source term represents the total spent fuel pool activity for 7-2/3 cores based on the spent fuel pool activity for 7-1/3 cores (1/3 of a core discharged over a period of 22 years plus 1/3 of the LOCA source term (presented in Table 1.7.6-1).

For Cask Drop Case 2, the source term represents the total spent fuel pool activity for 7-2/3 cores based on the spent fuel pool activity for 6-2/3 cores (1/3 of a core discharged over a period of 20 years plus 1/3 of the LOCA source term).

The Spent Fuel Cask Drop Source Terms are presented in Table 1.7.6-1.

1.8 Atmospheric Dispersion (*X/Q*) Factors

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 Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," December 2001, has been implemented. DG-1111 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 Stack to Control Room receptor point,
- Closest Main Steam Safety Valve (MSSV)/Atmospheric Dump Valve (ADV) to Control Room receptor point,
- Closest Feedwater Line (containment) penetration to Control Room receptor point,
- Closest Fuel Handling Building (FHB) wall to Control Room receptor point,
- Condenser to Control Room receptor point,
- RWT to Control Room receptor point, and
- Auxiliary Building louvers to Control Room receptor point.

Figure 1.8.1-1 provides a sketch of the general layout of St. Lucie Unit 1 that has been annotated to highlight the release and receptor point locations described above. 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 28° 41' 56".

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 unfavorable intake prior to intake isolation, the midpoint between the intakes during isolation, as well as values for the favorable intake due to the manual selection of the favorable control room intake after unisolation. These values include taking credit for dilution where allowed by DG-1111. Based on the layout of the site, the only cases that may take credit for dilution are when the releases are from the plant vent stack. However, dilution is not credited during the time period when the control room intakes are isolated for these cases.

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 during each of the three modes of control room ventilation.

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 are compared with the 5% overall site *X/Q* values for those boundaries, and the larger of the values are used in evaluations.

All of the releases are considered ground level releases because the highest possible release height is 184 feet (from the plant stack). 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 St. Lucie plant, the elevation of the top of the Unit 1 containment is given as 225.50 feet. 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 the ARCON96 onsite *X/Q* cases. This area of 1565 m^2 is calculated to be conservatively small in that the height used in the area calculation is from the highest roof elevation of a nearby building to the elevation of the bottom of the containment dome. The containment height used in the building wake term is the containment top elevation minus the bottom (grade) elevation of 19 feet. 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 10.0 meters. There are zero hours of calms in the joint frequency distribution data. This low number of calm hours is due to the positioning of the St. Lucie plant and its proximity to the Atlantic Ocean. The highest windspeed category is classified in RG 1.23, "Onsite Meteorological Programs," February 1972, as greater than 24 mph, however the PAVAN code requires that the maximum speed for each category be input. Therefore, the 30-mph value is chosen as the upper limit on the fastest windspeed category because the raw meteorological data showed that there were no hours with windspeeds faster than 30 mph.

1.8.3 Meteorological Data

Meteorological data over a five-year period (1996 through 2001) is used in the development of the new *X/Q* factors used in the analysis. The St. Lucie Plant, Unit No. 1, Meteorological Monitoring Program, complies with RG 1.23; "Onsite Meteorological Programs," 1972. The Meteorological Monitoring Program is described in Section 2.3.3 of the St. Lucie Plant Unit No. 1 UFSAR.

For the onsite *X/Q* determinations, the five years include the last six months of 1996; all of 1997, 1998, and 1999; the first six months of 2000; and all of 2001. The last six months of 2000 data are not included because of the poor quality of the raw data (i.e. significant portions of time with unrecorded data). Since the poor data period occurred in the middle of the time period under consideration, and that 5 years' worth of data is desired, the last six months of 1996 data are included at the beginning of the meteorological data file. For the offsite *X/Q* determinations, the five years are from 1997 through 2001.

Channel B is the primary channel for the temperature difference measurement data from the raw meteorological tower data for 1997-2000. For the 2001 data, channel A is taken as the primary channel for the temperature difference measurement data from the raw meteorological tower data.

The meteorological data is converted from the raw format into the proper formatting required to create the meteorological data files for the ARCON96 runs. Five years worth of meteorological data is used which meets the guidance set forth in Section 2.1 of DG-1111. The raw data for 1996 through 2001 was provided in electronic format in comma delimited text files. The data from these files were manipulated within a spreadsheet. The stability class for each hour data point is calculated using the ΔT method. The temperature difference is provided from two channels on the tower. Channel B is the primary channel and all the values for the temperature difference are channel B values for 1996 through 2000. However, if the channel B data is bad (indicated by a value of 99.99), then channel A data is used. Conversely, channel A is used as the primary channel for the 2001 data. The last 6 months of 2000 had an abundance of bad data. In order to obtain a more accurate representation of normal meteorological conditions at the St. Lucie site and to achieve 5 years worth of good data, only the last six months from the 1996 data are used and the last six months of 2000 are omitted.

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 St. Lucie Unit 1 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 43,454 and the number of missing data hours as 2,108. This yields a meteorological data recovery rate of 95.1%. No regulatory guidance is provided in DG-1111 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 95.1% 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 95.1% 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 St. Lucie site.

The meteorological data were also provided in annual joint frequency distribution format for 1997 through 2001. 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 percent of hours of that year that fell into each classification category. The data for each category (i.e. wind speed, wind direction, and stability class unique combination) were converted from percent to number of hours.

The number of hours for each classification is then rounded to the nearest whole hour. 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.

An additional process is performed on the met data used for the ARCON runs to determine the average air temperature swing over a 24-hour period for the five years' worth of data. The yearly data is combined so that the dates match the data used for the ARCON96 met file. That is the last 6 months of 1996 are included and the last 6 months of 2000 are omitted as previously explained. Any data determined to be invalid is excluded. The average air temperature range over the five years of meteorological data is calculated to be a 9.6°F temperature swing over any 24-hour period. A median value is also calculated. The median 24-hour period temperature swing value is 8.7°F. The higher value is used to support determining the leakage rate from the RWST.

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 15.4.1 of the UFSAR.

2.1.2 Compliance with RG 1.183 Regulatory Positions

The revised 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 6.2.6.1. 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 2.89 hr^{-1}. This removal is credited in the sprayed region prior to spray actuation and in the unsprayed region prior to and during spray actuation. Credit for elemental deposition is conservatively not credited after the time when credit for elemental removal by the sprays is terminated (at 2.065 hrs). A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region and in the sprayed region prior to spray actuation. No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 - Containment spray provides coverage to 86% of the containment. Therefore, the St. Lucie Unit 1 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. Based on the timing of the release phases provided in RG 1.183, Regulatory Position 3.3, Table 4, the expected maximum concentration should occur at 1.8 hours. The analysis confirmed this assumption. Based upon the conservatively assumed elemental iodine removal rate of 20 hr^{-1}, the DF of 200 is computed to occur

at 2.065 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 5.53 hr^{-1}, the DF of 50 is conservatively computed to occur at 2.507 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 St. Lucie Unit No. 1.

8. Regulatory Position 3.6 - Not applicable to St. Lucie Unit No. 1.

9. Regulatory Position 3.7 - A containment leak rate of 0.5% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.25% per day of the containment air.

10. Regulatory Position 3.8 - Routine containment purge is considered in this analysis.

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 120 seconds.

13. Regulatory Position 4.3 - SBVS is credited as being capable of maintaining the Shield Building Annulus at 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 Section 6.2. No exfiltration through the concrete wall of the Shield Building is expected to occur.

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

15. Regulatory Position 4.5 - 9.6% of the primary containment leakage is assumed to bypass the secondary containment. This bypass leakage is released from the plant stack via the RAB ventilation system without credit for filtration.

16. Regulatory Position 4.6 - The SBVS is credited as meeting the requirements of RG 1.52 and Generic Letter 99-02.

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 the value provided in Reference 5.16. The leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continue for the 30-day duration.

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

20. Regulatory Position 5.4 - The conservative calculated flashing fraction for the leaked ECCS sump liquid during recirculation is 6.1%; however, consistent with Regulatory Position 5.5, the flashing fraction for ECCS leakage is assumed to be 10%. This ECCS leakage enters the Reactor Auxiliary Building. For ECCS leakage back to the RWT, the analysis demonstrates

that the temperature of the leaked fluid will cool below 212°F prior to release into the tank.

21. Regulatory Position 5.5 - The amount of iodine that becomes airborne is conservatively assumed to be 10% of the total iodine activity in the leaked fluid for the ECCS leakage entering the Reactor Auxiliary Building. For the ECCS leakage back to the RWT, the sump and RWT pH history and temperature are used to evaluate the amount of iodine that enters the RWT air space.

22. Regulatory Position 5.6 - The temperature and pH history of the sump and RWT 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 RWT and filtration of the ECCS leakage release in the RAB as allowed by Regulatory Position 5.6.

23. Regulatory Position 6 - Not applicable to St. Lucie Unit No. 1.

24. Regulatory Position 7 - Containment purge is not considered as a means of combustible gas or pressure control in this analysis.

2.1.3 Methodology

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). 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 SIS signal is actuated when the appropriate setpoint 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.

<u>Release Inputs</u>
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.

Per TS 3.6.1.1 and 6.8.4.h (a), the leakage rate acceptance criteria for the containment is 0.5% of the containment air weight per day. Therefore, for the first 24 hours, the containment is assumed to leak at a rate of 0.5% 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 0.25% /day based on the post-LOCA primary containment pressure history.

The ESF leakage to the Auxiliary Building is assumed to be 4510 cc/hr, based upon two times the current licensing basis value of 2255 cc/hr. The leakage is conservatively assumed to start at 15 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that 10% of the

total iodine is released from the leaked liquid. However, a sump pH history evaluation demonstrates that the sump liquid will not contain significant amounts of elemental iodine during the time of interest for this analysis. Based upon the sump pH history, this analysis will conservatively assume that the chemical form of the iodine in the sump water is:

0 to 1 hour - 95 % aerosol, 4.85 elemental, and 0.15% organic, and

after 1 hour - 99.6% aerosol, 0.25% elemental, and 0.15% organic.

This analysis assumes that all of the elemental and organic iodine in the leaked fluid is volatile and becomes an airborne release to the ECCS area. Furthermore, all of the particulates in the 10% flashed fraction of the release are assumed to become airborne in the ECCS area.

The ECCS back-leakage to the RWT is assumed to be 2 gpm based upon doubling the specified 1 gpm leakage. The leakage is conservatively assumed to start at 15 minutes into the event and continue throughout the 30-day period. Based upon the sump pH history, this analysis conservatively assumes that the chemical form of the iodine in the sump water is:

0 to 1 hour - 95 % aerosol, 4.85 elemental, and 0.15% organic, and

after 1 hour - 99.6% aerosol, 0.25% elemental, and 0.15% organic.

However, when introduced into the acidic solution of the RWT inventory, there is a potential for the particulate iodine to convert back into the elemental form. Based upon the initial RWT pH of 4.5, the amount of iodine converted to the elemental form in the RWT is determined based upon the data provided in NUREG-5950. It is conservatively assumed that 2.5% of the particulate iodine is converted into the elemental form when it is leaked into the RWT. This conversion fraction is conservatively assumed to exist throughout the event even though the pH of the RWT is significantly neutralized during the course of the sump leakage. The model adds this regenerated elemental iodine to the RWT backleakage. The leakage is modeled as:

0 to 1 hour - 92.5% aerosol, 7.35% elemental, and 0.15% organic, and

after 1 hour - 97.1% aerosol, 2.75% elemental, and 0.15% organic.

The elemental iodine in the liquid leaked into the RWT is assumed to become volatile and partitioned between the liquid and vapor space in the RWT 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 RWT since no boiling occurs in the RWT. The release of the activity from the vapor space within the RWT is calculated based upon the displacement of air by the incoming leakage and the expansion and contraction due to the diurnal heating and cooling of the contents of the RWT tank. The adjusted release rate from the RWT presented in Table 2.1-3 equals the volume of air/vapor being released from the tank divided by the partition coefficient.

A 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-131and 100/E-bar gross activity) is available for release via this pathway. The release is conservatively modeled for 30 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 stack. The activity that bypasses the secondary containment is identified as being leaked within the RAB where it is collected by the ventilation system and released to the

environment via a ground level release from the plant stack without assumed filtration. The activity from the ECCS leakage into the RAB is modeled as a ground level release via the ECCS area ventilation system with a particulate removal efficiency of 99% and 97.5% efficiency for elemental and organic iodine. The activity from the RWT is modeled as an unfiltered ground level release from the RWT.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 580 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high containment pressure signal. A 50-second delay is applied to account for the time to reach the signal, the diesel generator start time, damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 580 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 580 cfm of unfiltered inleakage, and 1550 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, 97.5% elemental iodine, and 97.5% organic iodine.

<u>LOCA Removal Inputs</u>

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 2.89 hr^{-1}. This removal is credited in the sprayed region prior to spray actuation and in the unsprayed region prior to and during spray actuation. Credit for elemental deposition is conservatively not credited after the time when credit for elemental removal by the sprays is terminated (at 2.065 hrs). A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region and in the sprayed region prior to spray actuation. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 86% of the containment. Therefore, the St. Lucie Unit 1 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. Based on the timing of the release phases provided in RG 1.183, Regulatory Position 3.3, Table 4, the expected maximum concentration should occur at 1.8 hours. The analysis confirmed this assumption. Based upon the conservatively assumed elemental iodine removal rate of 20 hr^{-1}, the DF of 200 is conservatively computed to occur at 2.065 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 5.53 hr^{-1}, the DF of 50 is conservatively computed to occur at 2.507 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.

Leakage from containment that is collected by the shield building/secondary containment is processed by ESF filters prior to an assumed ground level release.

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 120 seconds.

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.

When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated at 50 seconds, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. 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 LOCA release points for the different modes of control room operation during the event.

For the EAB dose analysis, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period. The LPZ dose is determined using the *X/Q* factors for the appropriate time intervals. 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 five distinct activity releases:

1. Containment leakage via the secondary containment system.
2. Containment leakage bypassing the secondary containment
3. ESF system leakage into the Auxiliary Building.
4. ESF system leakage into the RWT.
5. 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 2.1-4, the sum of the results of all dose contributions 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 Handling Building (FHB) or inside of Containment. The FHA is described in Section 15.4.3 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 FHB release locations.

This analysis bounds dropping a fuel assembly either inside the containment (with the maintenance hatch open) or inside the FHB (without credit for filtration of the Fuel Handling Building exhaust). With the containment maintenance hatch open and filtration of the Fuel Handling Building exhaust not credited, the analyses are essentially identical for either the containment or the FHB release point except that the dispersion factors from the fuel handling building are slightly greater than the dispersion factors from the containment maintenance hatch. The source term released from the overlying water pool is the same for both the FHB 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.

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

Due to conflicting requirements outlined in Section 2.0 of Appendix B to Reg. Guide 1.183, the FHA cases were analyzed with elemental iodine decontamination factors of 500 and 285 (corresponding to an overall iodine decontamination factor of 200).

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 15.4.3.3.

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 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, a decontamination factor of 500 is applied to the elemental iodine and a decontamination factor of 1 is applied to the organic iodine. As a result, the breakdown of the iodine

species above the surface of the water is 57% elemental and 43% organic. Due to a conflicting requirement that the overall iodine decontamination factor be equal to 200 (which results in an elemental iodine decontamination factor of 285), an additional set of FHA cases were run with an elemental iodine decontamination factor of 285.

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 to the environment over a 2-hour period.

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

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

9. Regulatory Position 5.1 - The containment maintenance 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 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 maintenance hatch open or in the Fuel Handling Building without exhaust filtration. It is assumed that the fuel handling accident occurs at 72 hours after shutdown of the reactor per TS 3.9.3. 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. The activity released from the pool is 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 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 920 cfm of unfiltered fresh air and an assumed value of 1000 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 50-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 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% for elemental iodine, and 97.5% for 97.5% 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 maintenance hatch 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.

When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours.

For the EAB dose analysis, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period. The LPZ dose is determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are 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 either inside or outside of containment. Allowable fuel failure rates due to DNB and fuel centerline melt are determined for both break locations based upon the dose limits specified in Table 6 of RG 1.183. Depending on the location of the break, the affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to either the environment or the containment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cool down is achieved via the remaining unaffected SG. This event is described in the UFSAR, Section 15.4.6.

2.3.2 Compliance with RG 1.183 Regulatory Positions

The revised 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 - 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 1.7 is applied. The fraction of fission product inventory in the gap available for release due to DNB is consistent with Regulatory Position 3.2 and Table 3 of RG 1.183. For fuel centerline melt, the guidance provided in RG 1.183, Appendix H, Regulatory Position 1 is used to determine the release.

2. Regulatory Position 2 - Fuel damage is assumed for this event; therefore, iodine spike cases are not analyzed.

3. Regulatory Position 2.1 - Not applicable, fuel damage is assumed for this event.

4. Regulatory Position 2.2 - Not applicable, fuel damage is assumed for this event.

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 - 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. These fractions apply as a result of fuel damage.

7. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by proposed TS 3.4.6.2 (0.3 gpm total, 0.15 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.

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 12 hours. The release of radioactivity from the unaffected SG is assumed to continue until shutdown cooling is in operation and steam release from the SG is terminated.

10. Regulatory Position 5.4 - For the MSLB outside of containment all noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. For the MSLB inside of containment, all of the noble gas released from the primary system to the intact SG is assumed to be released directly to the environment and all of the noble gas released from the primary system to the faulted SG is assumed to be released directly to the containment.

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 (MSLB outside of containment) or the containment (MSLB inside of containment) with no mitigation. For the unaffected steam generator used for plant cooldown, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

14. Regulatory Position 5.5.2 - Any postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.

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

16. 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%. No reduction in the release is assumed from the faulted SG.

17. Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for the intact SG for St. Lucie Unit 1.

2.3.3 Other Assumptions

1. RG 1.183 does not address secondary coolant activity. 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. The steam mass release rates for the intact SG 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 SG is assumed to remain constant throughout the event.

5. For the MSLB outside of containment, 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 SG are postulated to occur from the MSSV or ADV with the most limiting atmospheric dispersion factors. For the MSLB inside of containment, releases from the affected SG are assumed to leak out of the containment via the same containment release points discussed for the LOCA in Section 2.1.

6. For the MSLB inside of containment, natural deposition of the radionuclides is credited. Containment sprays are not credited.

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 consists of two cases; one case assumes a double-ended break of one main steam line outside of containment, and the second case assumes a double-ended break of one main steam line inside of containment. Upon a MSLB, the affected SG rapidly depressurizes and releases the initial contents of the SG to the environment (or containment). The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the remaining unaffected SG. The analysis assumes that the entire fluid inventory from the affected SG is immediately released to the environment (or containment). Additional activity, based on the TS 3.4.6.2 primary-to-secondary leakage limits (SG tube leakage), is released via the unaffected SG via system relief valves until the Shutdown Cooling (SDC) system is placed in operation to continue heat removal from the primary system. Primary coolant is also released into the affected steam generator by leakage across the SG tubes. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DE I-131 permitted by TS 3.7.1.4. Activity is released to the environment (or containment) 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 faulted steam generator is completely isolated at 12 hours (primary system temperature less than 212°F). Primary-to-secondary tube leakage is also postulated to occur in the unaffected SG. Activity is released via steaming from the unaffected SG MSSVs/ADVs until the decay heat generated in the reactor core can be removed by the SDC system 8 hours into the MSLB event. These release assumptions are consistent with the requirements of RG 1.183.

Allowable levels of fuel failure for DNB and fuel centerline melt are determined for both the MSLB outside of containment and the MSLB inside of containment. These allowable fractions are based on the dose limits specified in Table 6 of RG 1.183. The activity released from the fuel that is assumed to experience DNB is based on Regulatory Positions 3.1, 3.2, and Table 3 of RG 1.183. The activity released from the fuel that is assumed to experience fuel centerline melt is based on Regulatory Position 1 of Appendix H to RG 1.183.

A radial peaking factor of 1.70 is applied in the development of the source terms.

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 920 cfm of unfiltered fresh air and an assumed value of 580 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 50-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 0 cfm of makeup flow from the outside, 580 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 580 cfm of unfiltered inleakage, and 1550 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, 97.5% for elemental iodine, and 97.5% for 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.

For both the MSLB outside of containment and inside of containment, releases from the intact SG are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs*. When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours.

For the MSLB outside of containment, releases from the faulted SG are assumed to occur from the steam line that produces the most limiting *X/Qs*. When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. For the MSLB inside of containment, the *X/Qs* for containment leakage are assumed to be identical to those for the LOCA discussed in Section 2.1.

For the EAB dose analysis, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period. The LPZ dose is determined using the *X/Q* factors for the appropriate time intervals. 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 inside and outside of containment with DNB and FCM fuel failure are analyzed. As shown in Table 2.3-3, the results of all four 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 St. Lucie Unit No. 1 SGTR event. This event is described in Section 15.4.4 of the UFSAR. The ruptured tube flow rate and cooldown steam releases are conservatively assumed for the St. Lucie Unit 1 replacement steam generators.

2.4.2 Compliance with RG 1.183 Regulatory Positions

The revised 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 - 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.

2. Regulatory Position 2 - No fuel damage is postulated to occur for the St. Lucie Unit No. 1 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 TS 3.4.8, Fig. 3.4-1 value of 60.0 μCi/gm DE I-131. 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 TS 3.4.8 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 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 proposed change to TS 3.4.6.2 (0.3 gpm total, 0.15 to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.

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 12 hours. The release of radioactivity from the unaffected SG is assumed to continue until shutdown cooling is in operation and steam release from the SG is terminated. The current Licensing Basis for the termination of the affected SG activity release states that the affected SG is isolated within 30 minutes by operator action. This isolation terminates releases from the affected SG, while primary-to-secondary leakage continues to provide activity for release from the unaffected SG.

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 - Both steam generators effectively maintain tube coverage. Therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

    - 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 St. Lucie Unit 1.

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." The iodine spike activity appearance rates are provided in Table 2.4-5.

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 loss of power concurrent with the reactor trip at 329 seconds, the loss of circulating water through the condenser would 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 ADVs or MSSVs. This direct steam relief continues until the faulted steam generator is isolated. The isolation is assumed to require 30 minutes. The SG is isolated on the secondary side by closing the associated inlet and outlet valves.

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 unaffected SG via the ADVs until the heat removal system is placed in operation to continue heat removal from the primary system.

Per the St. Lucie Unit 1 UFSAR, Section 15.4.4.5.1, no fuel failure 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 TS 3.4.8 (see 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 until the ruptured steam generator is isolated at 30 minutes. The unaffected SG is used to cool down the plant during the SGTR event. Primary-to-secondary tube leakage is also postulated into the intact SG. Activity is released via steaming from the unaffected SG ADVs until the decay heat generated in the reactor core can be removed by the Shutdown Cooling (SDC) 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 TS 3.4.8. 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. 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 three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 1000 cfm of unfiltered inleakage.

- After the start of the event, the Control Room isolation is initiated on a CR intake radiation monitor signal, which is set at 2 times background. This assures that the CR isolation is initiated prior to any significant activity being introduced into the CR ventilation system. For this event, it is conservatively assumed that the CR isolation signal is delayed until the release from the ADVs/MSSVs is initiated at 329.4 seconds. An additional 50-second delay is applied to account for the diesel generator start time, fan start and damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% for elemental iodine, and 97.5% for organic iodine.

2.4.5 Radiological Consequences

The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. Releases from the faulted SGs are postulated from the nearest safety relief valve to the Control Room. Therefore, the closest ADV/MSSV – Control Room release-receptor combination is assumed. When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated at 50 seconds, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. 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 SGTR release points for the different modes of Control Room operation during the event.

For the EAB dose analysis, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period. The LPZ dose is determined using the *X/Q* factors for the appropriate time intervals. 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 TS 3.4.8 limits, are analyzed. As shown in Table 2.4-6, the radiological consequences of the St. Lucie Unit 1 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 ADVs 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 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 1.7 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 proposed Technical Specification 3.4.6.2, as 0.3 gpm total and 0.15 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 St. Lucie Unit 1. Therefore, the iodine and transport model for release from the SGs is as follows:

    - Appendix E, Regulatory Position 5.5.1 - Both 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 St. Lucie Unit 1.

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. 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.

3. RG 1.183 does not address secondary coolant equilibrium specific activity for the alkali metals. This analysis assumes that the equilibrium specific activity of the alkali metals resulting from primary-to-secondary leakage into the SGs is assumed to be 10% of the primary coolant equilibrium concentration.

4. This analysis assumes that the DNB fuel damage is limited to 13.7% breached fuel assemblies.

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 13.7% of the fuel assemblies are assumed damaged. A radial peaking factor of 1.7 is assumed. The activity released from the fuel 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 ADVs and MSSVs until the decay heat generated in the reactor core can be removed by the shutdown cooling 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 920 cfm of unfiltered fresh air and an assumed value of 1000 cfm of unfiltered inleakage.

- After the start of the event, the Control Room isolation is initiated on a CR intake radiation monitor signal, which is set at 2 times background. This low setpoint assures that the CR isolation will be initiated prior to any significant activity being introduced into the CR ventilation system. For this event, it is assumed that the CR isolation signal will coincide with the assumed release from the MSSVs. An additional 50-second delay is applied to account for the diesel generator start time, fan start and damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% for elemental iodine, and 97.5% 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 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.

The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated at 50 seconds, the limiting *X/Q* corresponds to the midpoint between the two Control Room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. 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 Locked Rotor release points for the different modes of Control Room operation during the event.

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. For the EAB dose analysis, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period.

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 Control Element Assembly Ejection (CEA)

2.6.1 Background

This event consists of an uncontrolled withdrawal of a single control element assembly (CEA). This event is the same as the Rod Ejection event referred to in RG 1.183. The CEA 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/ADVs. Two CEA 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 15.4.5.

2.6.2 Compliance with RG 1.183 Regulatory Positions

The CEA 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 1.7 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. Containment sump pH is controlled to 7.0 or higher.

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 shield building ventilation system (SBVS) 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

0.5% for the first 24 hours and 0.25% 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 (0.3 gpm total, 0.15 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 - Both 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 St. Lucie Unit 1.

2.6.3 Other Assumptions

1. RG 1.183 does not address secondary coolant activity. This analysis assumed 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. 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 ADV with the most limiting atmospheric dispersion factors. For the CEA 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 CEA 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 shield building ventilation system (SBVS) 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/ADVs until the decay heat generated in the reactor core can be removed by the Shutdown Cooling (SDC) 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. 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 CEA Ejection is evaluated with the assumption that 0.5% of the fuel experiences FCM and 9.5% 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 1.70 is applied in the development of the source terms.

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 920 cfm of unfiltered fresh air and an assumed value of 1000 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 50-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 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% for elemental iodine, and 97.5% 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 operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

For the CEA secondary side release case, releases from the SGs are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs*. When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the

ventilation system is isolated, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. For the CEA 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 dose analysis, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period. The LPZ dose is determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

The radiological consequences of the CEA 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 Inadvertent Opening of a Main Steam Safety Valve (IOMSSV):

2.7.1 Background

This event is caused by an Inadvertent Opening of a Steam Generator MSSV. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products contained in the primary coolant before the accident are discharged from the primary into the secondary system. The analysis assumes that the SG tubes do not remain covered and therefore no credit is taken for scrubbing in the SG or any credit for a flashing fraction for the primary leakage into the SGs. As a result, all of this leaked RCS radioactivity is released to the outside atmosphere from the secondary coolant system through the steam generator via the MSSVs. In addition, all of the activity initially present in the SGs is assumed to be released to the environment over a 2-hour period. Radiological releases due to the opening of a power operated atmospheric dump valve are bounded by the stuck open MSSV event. This IOMSSV event is described in Section 15.2.11.3.2 of the UFSAR.

2.7.2 Compliance with RG 1.183 Regulatory Positions

Since Regulatory Guide 1.183 does not provide specific guidance for this event the guidance of Appendix G for the RCP Shaft Seizure (Locked Rotor) event is judged to be closely applicable to the conditions of a stuck open MSSV. Therefore the following discussion for the IOMSSV refers to the RG 1.183 positions as stated in Appendix G for the Locked Rotor event.

1. Regulatory Position 1 - No fuel damage is postulated to occur. The source term for this event is due to the initial RCS and Secondary side activity present at the beginning of the event.

2. Regulatory Position 2 - No fuel damage is assumed for this event.

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

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

5. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the steam generators, as specified by proposed Technical Specification 3.4.6.2, as 0.3 gpm total and 0.15 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 the RCS is cooled to 212 F and the primary-to-secondary leakage is 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 not assumed to remain covered throughout this event for St. Lucie Unit 1. Therefore, the iodine and transport model for release from the SGs is as follows:

- Appendix E, Regulatory Position 5.5.1 - Both steam generators are assumed to "dryout." Therefore, all of the primary-to-secondary leakage is assumed to flash to steam and be released to the environment with no mitigation.

- Appendix E, Regulatory Position 5.5.2 - All 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 flash for this event.

- Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water in the SGs is assumed to be released directly to the environment over a 2-hour period.

- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is postulated for this event for St. Lucie Unit 1.

2.7.3 Other Assumptions

The initial RCS activity is assumed to be at the TS 3.4.8 limit of 1.0 µCi/gm Dose Equivalent I-131. 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.

2.7.4 Methodology

Input assumptions used in the dose consequence analysis of the Inadvertent Opening of a MSSV event are provided in Table 2.7-1. Primary coolant is released to the SGs as a result of postulated primary-to-secondary leakage. The activity in the RCS tube leakage is released to the atmosphere via steaming from the open MSSVs until the RCS is cooled to 212°F at 12 hours into the event. These release assumptions are consistent with the guidance of RG 1.183. In addition, the entire secondary side activity is released to the environment over a 2-hour period.

The IOMSSV event is evaluated with the assumption that the activity released from the primary and secondary coolant is established at the Technical Specification limit of 1.0 µCi/gm Dose Equivalent I-131. 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.

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 920 cfm of unfiltered fresh air and an assumed value of 1000 cfm of unfiltered inleakage.

- After the start of the event, the Control Room isolation is initiated on a CR intake radiation monitor signal, which is set at 2 times background. This low setpoint assures that the CR isolation will be initiated prior to any significant activity being introduced into the CR ventilation system. For this event, it is assumed that the CR isolation signal will coincide with the assumed release from the MSSVs. An additional 50-second delay is applied to account for the diesel generator start time, fan start and damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of 450 cfm of

filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% elemental iodine, and 97.5% 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 operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3.

The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated at 50 seconds, the limiting *X/Q* corresponds to the midpoint between the two Control Room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. The *X/Q* values for the various combinations of release points and receptor locations are presented in Table1.8.1-2. Table 1.8.1-3 presents the Release-Receptor pairs applicable to the Control Room dose from the MSSV release points for the different modes of Control Room operation during the event.

The EAB and LPZ dose consequences are determined using the *X/Q* factors provided Table 1.8.2-1 for the appropriate time intervals. For the EAB dose calculation, the *X/Q* factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour *X/Q* factor provides a more conservative determination of the EAB dose, because the *X/Q* factor for this time period is higher than for any other time period.

RG 1.183 lists no specific acceptance criteria for this event; therefore, the most limiting dose limits are used. Per Section 4.4 and Table 6 of RG 1.183, the most limiting dose limits for the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) 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)

*Control room dose limit is specified in 10CFR50.67

The radiological consequences of the Inadvertent Opening of the MSSV event are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.7-2, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.8 Spent Fuel Cask Drop

2.8.1 Background

The purpose of this analysis is to reanalyze the radiological consequences of the cask drop accident presented in Section 9.1.4.3 of the St. Lucie Unit 1 UFSAR. RG 1.183 does not provide any specific guidance for the cask drop event; therefore, the requirements of Appendix B of the RG (fuel handling accident) are followed for the cask drop reanalysis.

Section 9.1.4.3 of the UFSAR describes two cask drop cases:

Case 1
"One-third of a core is placed in the spent fuel pool each year during refueling for the next 23 years, until the pool is filled (7 2/3 cores). The number of assemblies damaged is equal to the number offloaded during a normal refueling plus the remainder of the pool filled with discharged assemblies from the previous refuelings."

Per Tech. Spec. 3.9.14, 1180 hours of decay is required before movement of the spent fuel cask for Case 1.

Case 2
"One-third of a core is placed in the spent fuel pool each year during refueling for the next 20 years. Following the 21st year of operation, the entire core is removed from the reactor and placed into the pool, which fills the pool (7 2/3 cores). The number of assemblies damaged is equal to a full-core offload plus the remainder of the pool filled with discharged assemblies from the previous refuelings."

Per Tech. Spec. 3.9.14, 1490 hours of decay is required before movement of the spent fuel cask for Case 2.

Due to conflicting requirements outlined in Section 2.0 of Appendix B to Reg. Guide 1.183, the cask drop cases were analyzed with elemental iodine decontamination factors of 500 and 285 (corresponding to an overall iodine decontamination factor of 200).

2.8.2 Compliance with RG 1.183 Regulatory Positions

The Cask Drop 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 the maximum amount of fuel potentially stored in the Fuel Pool.

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 1.7.6 provides a discussion of how the Cask Drop source term is developed. A listing of the source terms is provided in Table 1.7.6-1. The gap activity available for release is specified by Table 3 of RG 1.183. This activity is assumed to be released 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 assemblies. Therefore, a decontamination factor of 500 is applied to the elemental iodine and a decontamination factor of 1 is applied to the organic iodine. As a result, the breakdown of the iodine species above the surface of the water is 57% elemental and 43% organic. Due to a conflicting requirement that the overall iodine decontamination factor be equal to 200 (which results in an elemental iodine decontamination factor of 285), an additional set of cask drop cases were run with an elemental iodine decontamination factor of 285.

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 radioactive material released from the fuel pool is assumed to be released from the building to the environment within 2 hours.

7. Regulatory Position 4.2 - No filtration is assumed.

8. Regulatory Position 4.3 - No dilution is assumed.

9. Regulatory Position 5 - The event does not occur in the containment.

2.8.3 Other Assumptions

The dose acceptance criteria for the Spent Fuel Cask Drop are assumed to be the same as those for the Fuel Handling Accident.

2.8.4 Methodology

The input assumptions used in the dose consequence analysis of the Cask Drop are provided in Table 2.8-1. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from all of the fuel assemblies 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 is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel are assumed to be retained by the pool. The iodine released from the damaged fuel is assumed to be composed of 99.85% elemental and 0.15% organic. The activity released from the pool is assumed to leak to the environment over a two-hour period.

The source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. Section 1.7.6 discusses the development of the Cask Drop source terms for both cases, which are listed in Table 1.7.6-1.

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 920 cfm of unfiltered fresh air and an assumed value of 1000 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 50-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 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to

the control room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% elemental iodine, and 97.5% organic iodine.

2.8.5 Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the location of the closest corner FHB 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.

When the Control Room Ventilation System is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours.

For the EAB & LPZ dose calculation, the *X/Q* factors from Table 1.8.2-1 are assumed.

The radiological consequences of the Spent Fuel Cask Drop are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.8-2, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.9 Waste Gas Decay Tank Rupture

2.9.1 Background

This event involves a major rupture of one of the Waste Gas Decay Tanks as currently presented in Section 15.4.2 of the St. Lucie Unit 1 UFSAR. This analysis assumes that the ruptured WGDT contains an inventory equivalent to the tank activity limit specified in the of the St. Lucie Unit 1 Technical Specification 3.11.2.6. The entire source term is applied to this component at the beginning of the event. The leak rate from the WGDT to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the tank contents 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 Ventilation 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 Waste Gas Decay Tank Rupture. Therefore, this analysis will rely primarily upon the current UFSAR licensing basis for guidance on performance of this event.

2.9.3 Other Assumptions

The St. Lucie Unit 1 Waste Gas Decay Tank source term is assumed to be the same as that for the St. Lucie Unit No. 2 WGDT. The source term provided in St. Lucie Unit 2 UFSAR Table 15.7.4.1-2 is used to provide a representative isotopic distribution for the activity in the WGDT. This activity is based upon the conditions specified in St. Lucie Unit 2 UFSAR Table 15.7.4.1-1. That table is based upon the plant operating at the power level of 2700 MW_{th} with one percent failed fuel for an extended period of time sufficient to achieve equilibrium radioactive concentrations in the Reactor Coolant System. As soon as possible following shutdown, all noble gases are removed from the RCS and transferred to the WGDT. Radioactive decay is assumed only for the minimum period of time required to transfer the gases to the gas decay tank. The WGDT noble gas isotopic inventory specified in St. Lucie Unit 2 UFSAR Table 15.7.4.1-2 is scaled up by a factor of 24.84 to satisfy the TS 3.11.2.6 limit of 285,000 curies of noble gases (considered as Xe-133). The scaled up WGDT source term is provided in Table 2.9-2.

2.9.4 Methodology

The dose assessment model releases the above-prescribed inventory from the tank at a high rate of release to simulate the tank rupture. The contents are released to the environment without any hold up, dilution or filtration.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 1000 cfm of unfiltered inleakage.
- After the start of the event, the Control Room isolation is initiated on a CR intake radiation monitor signal, which is set at 2 times background. This low setpoint assures that the CR isolation is initiated prior to any significant activity being introduced into the CR ventilation system. For

this event, it is assumed that the CR isolation signal will coincide with the assumed release from the Waste Gas Decay Tank. An additional 50-second delay is applied to account for the diesel generator start time, fan start and damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 1000 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the Control Room. During this operational mode, the air flow distribution consists of 450 cfm of filtered makeup flow, 1000 cfm of unfiltered inleakage, and 1550 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, 97.5% for elemental iodine, and 97.5% for organic iodine.

2.9.5 Radiological Consequences

The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room intake. When the Control Room ventilation system is in normal mode, the most limiting *X/Q* corresponds to the worst air intake to the control room. When the ventilation system is isolated at 50 seconds, the limiting *X/Q* corresponds to the midpoint between the two Control Room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. 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 WGDT release points for the different modes of Control Room operation during the event.

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

The Waste Gas Decay Tank Rupture is classified in the St. Lucie Unit 1 UFSAR Table 15.1.1-1 as a Class 3 event (same classification as the Fuel Handling Accident). The acceptable dose limits for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are identified in UFSAR Table 15.4.1-5 as "well within the 10CFR100." Although RG 1.183 does not provide specific acceptance criteria for a Waste Gas Decay Tank Rupture, it does provide guidance for a Fuel Handling Accident that St. Lucie Unit 1 categorized as the same Class 3 event. Therefore, the RG 1.183 acceptance criteria for the FHA will also be applied to the WGDT Rupture. The control room dose limits are specified in 10CFR50.67. Therefore the dose limits are:

Area	Dose Criteria	
EAB	6.3 rem TEDE	(for the worst two hour period)
LPZ	6.3 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the Waste Gas Decay Tank Rupture event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 2.9-3, "WGDT Dose Consequences," the radiological consequences of the Waste Gas Decay Tank Rupture are all within the appropriate acceptance criteria.

2.10 Environmental Qualification (EQ)

The St. Lucie Unit No. 1 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 St. Lucie Unit No. 1 EQ analyses will continue to be based on TID-14844 assumptions.

3.0 Summary of Results

Results of the St. Lucie Unit 1 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 analysis has been made 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), Control Element Assembly (CEA) Ejection, Inadvertent Opening of a Main Steam Safety Valve (IOMSSV), Spent Fuel Cask Drop, and Waste Gas Decay Tank (WGDT) Rupture have been made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the St. Lucie Unit No. 1 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 580 cfm.

5.0 References

5.1 St. Lucie Unit No. 1 Updated Final Safety Analysis Report (UFSAR), (through Amendment 18).

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

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

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

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

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

5.7 Florida Power & Light Company, St. Lucie Unit No. 1 Technical Specifications (through Amendment 186).

5.8 PSL-ENG-SENS-03-022, "Engineering Evaluation, Containment Bypass Leakage History, St. Lucie Nuclear Plant, Units 1 & 2," Revision 0, Florida Power & Light Company.

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 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).

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

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

5.14 "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).

5.15 Numerical Applications Inc., NAI-9912-04, Revision 2, "RADTRAD-NAI Version 1.0p3(QA) Documentation," July 2002.

5.16 PSL-ENG-SENS-03-001, "Engineering Evaluation, Alternate Source Term Design Inputs, St. Lucie Nuclear Plant, Unit 1," Revision 1, Florida Power & Light Company.

5.17 Florida Power & Light Company, St. Lucie Unit No. 1 Core Operating Limits Report (COLR), Cycle 17, Revision 0.

5.18 NRC Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," December 2001. (Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" was issued during the final development of this Technical Report. At that point, the associated calculations and other documentation supporting this report were already reviewed and approved. RG 1.194 was reviewed subsequently and found to be consistent with the portions of DG-1111 that were utilized for the project.)

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

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

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

5.22 St. Lucie Unit No. 2 Updated Final Safety Analysis Report (UFSAR), (through Amendment 14).

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

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 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.

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

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

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

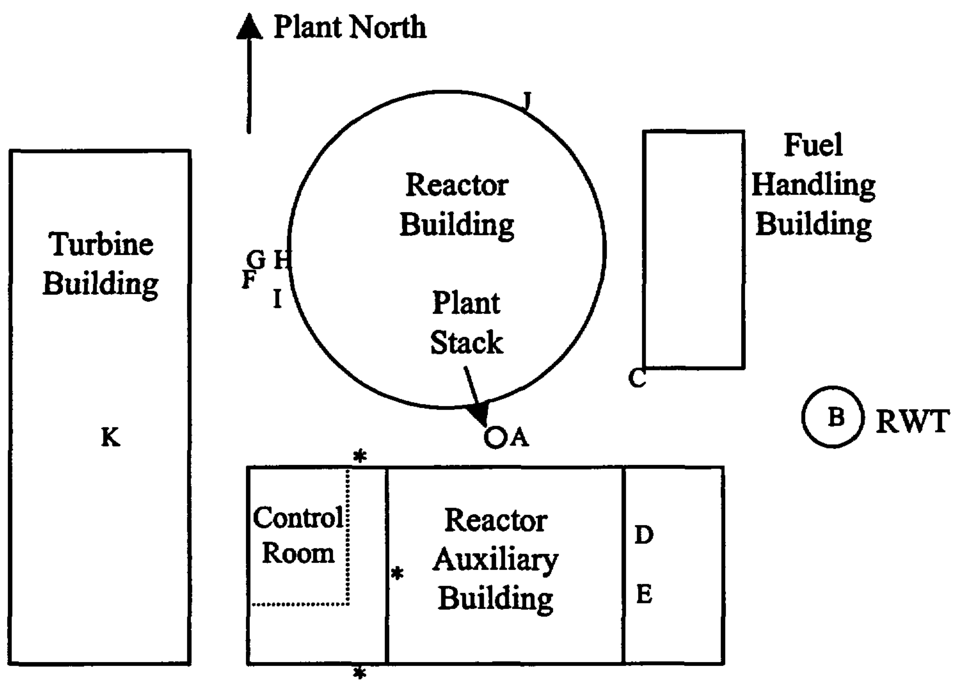
5.30 NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992.

5.31 Kewaunee Nuclear Power Plant AST submittal dated March 19, 2002 and subsequent Issuance of Amendment (IA) and Safety Evaluation (SE) issued March 17, 2003.

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

Figure 1.8.1-1

Onsite Release-Receptor Location Sketch



(Not to scale)

* – Control Room Intakes / Receptor Point
A – Plant Stack
B – RWT
C – FHB Closest Point
D – Louver L-7B
E – Louver L-7A
F – Closest ADV
G – Closest MSSV
H – Closest Main Steam Line Point (containment penetration)
I – Closest Feedwater Line Point (containment penetration)
J – Containment Maintenance Hatch
K – Condenser

Table 1.6.3-1

Control Room Ventilation System Parameters

Parameter	**Value**
Control Room Volume	62,700 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	920 cfm
Unfiltered Inleakage (Total) Non-LOCA/MSLB LOCA and MSLB	 1000 cfm 580 cfm
Emergency Operation	
Isolation Mode:	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	2000 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage (Total) Non-LOCA/MSLB LOCA and MSLB	 1000 cfm 580 cfm
Filtered Make-up Mode:	
Filtered Make-up Flow Rate	450 cfm
Filtered Recirculation Flow Rate	1550 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage (Total) Non-LOCA/MSLB LOCA and MSLB	 1000 cfm 580 cfm
Filter Efficiencies	
Particulate	99%
Elemental	97.5%
Organic	97.5%

Table 1.6.3-2

LOCA Direct Shine Dose

Source	Direct Shine Dose (rem)
Containment	0.025
Filters	0.074
External Cloud	0.069
Total	0.168

The total direct shine dose to a control room operator for the 30-day period following a LOCA is conservatively rounded to a minimum of 0.2 rem for use in calculating the total Control Room dose for each event.

Table 1.7.2-1
Primary Coolant Source Term

Nuclide	μCi/gm	Nuclide	μCi/gm
I-131	0.7920	SR-90	4.551E-04
I-132	0.2175	CR-51	6.627E-03
I-133	1.1293	FE-59	3.715E-05
I-134	0.1237	CO-60	9.051E-04
I-135	0.5387	SR-91	6.208E-03
H-3	2.302E-01	Y-90	1.779E-03
KR-85M	2.599E+00	Y-91	1.936E-01
KR-85	1.543E+00	ZR-95	1.630E-06
KR-87	1.413E+00	MO-99	3.541E+00
KR-88	4.534E+00	RU-103	7.203E-03
RB-88	4.447E+00	RU-106	4.326E-04
RB-89	1.116E-01	TE-129	4.377E-02
XE-131M	2.582E+00	TE-132	5.755E-01
XE-133	3.156E+02	TE-134	4.568E-02
XE-135	1.313E+01	BA-140	1.065E-02
BR-84	8.126E-02	LA-140	1.020E-02
CS-134	1.744E-01	CE-144	7.203E-03
CS-136	4.447E-02	PR-143	1.018E-02
CS-137	5.580E-01	MN-54	4.796E-05
CS-138	1.203E+00	CO-58	8.126E-03
SR-89	8.840E-03		

Table 1.7.3-1
Secondary Side Source Term

Isotope	µCi/gm
I-131	0.07920
I-132	0.02175
I-133	0.11293
I-134	0.01237
I-135	0.05387

Table 1.7.4-1
LOCA Containment Leakage Source Term

Nuclide	Curies	Nuclide	Curies
Co-58	0.000E+00	Pu-239	3.828E+04
Co-60	0.000E+00	Pu-240	6.456E+04
Kr-85	1.152E+06	Pu-241	1.626E+07
Kr-85m	1.784E+07	Am-241	2.152E+04
Kr-87	3.383E+07	Cm-242	6.998E+06
Kr-88	4.752E+07	Cm-244	1.053E+06
Rb-86	2.348E+05	I-130	4.626E+06
Sr-89	6.480E+07	Kr-83m	8.634E+06
Sr-90	9.253E+06	Xe-138	1.198E+08
Sr-91	8.105E+07	Xe-131m	8.582E+05
Sr-92	8.882E+07	Xe-133m	4.765E+06
Y-90	9.615E+06	Xe-135m	3.081E+07
Y-91	8.483E+07	Cs-138	1.334E+08
Y-92	8.925E+07	Cs-134m	5.846E+06
Y-93	1.046E+08	Rb-88	4.841E+07
Zr-95	1.206E+08	Rb-89	6.176E+07
Zr-97	1.207E+08	Sb-124	2.157E+05
Nb-95	1.220E+08	Sb-125	1.797E+06
Mo-99	1.405E+08	Sb-126	1.244E+05
Tc-99m	1.230E+08	Te-131	6.773E+07
Ru-103	1.320E+08	Te-133	8.797E+07
Ru-105	1.010E+08	Te-134	1.188E+08
Ru-106	6.560E+07	Te-125m	3.947E+05
Rh-105	9.303E+07	Te-133m	5.267E+07
Sb-127	9.609E+06	Ba-141	1.184E+08
Sb-129	2.678E+07	Ba-137m	1.216E+07
Te-127	9.546E+06	Pd-109	3.771E+07
Te-127m	1.294E+06	Rh-106	7.109E+07
Te-129	2.637E+07	Rh-103m	1.189E+08
Te-129m	3.930E+06	Tc-101	1.293E+08
Te-131m	1.151E+07	Eu-154	1.606E+06
Te-132	1.073E+08	Eu-155	1.088E+06
I-131	7.686E+07	Eu-156	2.847E+07
I-132	1.094E+08	La-143	1.086E+08
I-133	1.486E+08	Nb-97	1.218E+08
I-134	1.616E+08	Nb-95m	8.606E+05

Nuclide	Curies	Nuclide	Curies
I-135	1.396E+08	Pm-147	1.187E+07
Xe-133	1.492E+08	Pm-148	2.220E+07
Xe-135	4.333E+07	Pm-149	4.726E+07
Cs-134	2.606E+07	Pm-151	1.686E+07
Cs-136	7.018E+06	Pm-148m	2.843E+06
Cs-137	1.284E+07	Pr-144	1.021E+08
Ba-139	1.307E+08	Pr-144m	1.218E+06
Ba-140	1.258E+08	Sm-153	5.086E+07
La-140	1.310E+08	Y-94	1.062E+08
La-141	1.190E+08	Y-95	1.152E+08
La-142	1.146E+08	Y-91m	4.705E+07
Ce-141	1.208E+08	Br-82	6.291E+05
Ce-143	1.094E+08	Br-83	8.606E+06
Ce-144	1.014E+08	Br-84	1.470E+07
Pr-143	1.088E+08	Am-242	1.041E+07
Nd-147	4.809E+07	Np-238	5.399E+07
Np-239	1.960E+09	Pu-243	6.043E+07
Pu-238	5.475E+05		

Table 1.7.5-1
Fuel Handling Accident Source Term

Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)
Co-58	0.000E+00	I-135	1.094E+06	Sb-126	9.746E+02
Co-60	0.000E+00	Xe-133	1.169E+06	Te-131	5.306E+05
Kr-85	9.025E+03	Xe-135	3.395E+05	Te-133	6.892E+05
Kr-85m	1.398E+05	Cs-134	2.042E+05	Te-134	9.307E+05
Kr-87	2.650E+05	Cs-136	5.498E+04	Te-125m	3.092E+03
Kr-88	3.723E+05	Cs-137	1.006E+05	Te-133m	4.126E+05
Rb-86	1.839E+03	Ba-139	1.024E+06	Ba-141	9.276E+05
Sr-89	5.076E+05	Ba-140	9.855E+05	Ba-137m	9.526E+04
Sr-90	7.249E+04	La-140	1.026E+06	Pd-109	2.954E+05
Sr-91	6.350E+05	La-141	9.323E+05	Rh-106	5.569E+05
Sr-92	6.958E+05	La-142	8.978E+05	Rh-103m	9.315E+05
Y-90	7.532E+04	Ce-141	9.464E+05	Tc-101	1.013E+06
Y-91	6.646E+05	Ce-143	8.571E+05	Eu-154	1.258E+04
Y-92	6.992E+05	Ce-144	7.944E+05	Eu-155	8.524E+03
Y-93	8.194E+05	Pr-143	8.524E+05	Eu-156	2.230E+05
Zr-95	9.448E+05	Nd-147	3.767E+05	La-143	8.508E+05
Zr-97	9.456E+05	Np-239	1.535E+07	Nb-97	9.542E+05
Nb-95	9.558E+05	Pu-238	4.758E+03	Nb-95m	6.742E+03
Mo-99	1.101E+06	Pu-239	2.999E+02	Pm-147	9.299E+04
Tc-99m	9.636E+05	Pu-240	5.058E+02	Pm-148	1.739E+05
Ru-103	1.034E+06	Pu-241	1.274E+05	Pm-149	3.702E+05
Ru-105	7.912E+05	Am-241	1.686E+02	Pm-151	1.321E+05
Ru-106	5.139E+05	Cm-242	5.482E+04	Pm-148m	2.227E+04
Rh-105	7.288E+05	Cm-244	1.516E+04	Pr-144	7.999E+05
Sb-127	7.528E+04	I-130	3.624E+04	Pr-144m	9.542E+03
Sb-129	2.098E+05	Kr-83m	6.764E+04	Sm-153	3.984E+05
Te-127	7.478E+04	Xe-138	9.385E+05	Y-94	8.320E+05
Te-127m	1.014E+04	Xe-131m	6.723E+03	Y-95	9.025E+05
Te-129	2.066E+05	Xe-133m	3.733E+04	Y-91m	3.686E+05
Te-129m	3.079E+04	Xe-135m	2.414E+05	Br-82	4.928E+03
Te-131m	9.017E+04	Cs-138	1.045E+06	Br-83	6.742E+04
Te-132	8.406E+05	Cs-134m	4.580E+04	Br-84	1.152E+05
I-131	6.021E+05	Rb-88	3.792E+05	Am-242	8.155E+04
I-132	8.571E+05	Rb-89	4.838E+05	Np-238	4.289E+05
I-133	1.164E+06	Sb-124	1.690E+03	Pu-243	4.734E+05
I-134	1.266E+06	Sb-125	1.408E+04		

Table 1.7.6-1
Spent Fuel Cask Drop Source Terms

	Case 2 - Source Term			Case 1 - Source Term			
Nuclide	LOCA Source (Curies)	20 Years (6 2/3 Cores) (Curies)	Total (Curies)	LOCA Source (Curies)	1/3 LOCA Source (Curies)	22 Years (7 2/3 Cores) (Curies)	Total (Curies)
Co-58	0.000E+00	0.000E+00	**0.000E+00**	0.000E+00	0.000E+00	0.000E+00	**0.000E+00**
Co-60	0.000E+00	0.000E+00	**0.000E+00**	0.000E+00	0.000E+00	0.000E+00	**0.000E+00**
Kr-85	1.152E+06	4.173E+06	**5.325E+06**	1.152E+06	3.840E+05	4.365E+06	**4.749E+06**
Kr-85m	1.784E+07	0.000E+00	**1.784E+07**	1.784E+07	5.947E+06	0.000E+00	**5.947E+06**
Kr-87	3.383E+07	0.000E+00	**3.383E+07**	3.383E+07	1.128E+07	0.000E+00	**1.128E+07**
Kr-88	4.752E+07	0.000E+00	**4.752E+07**	4.752E+07	1.584E+07	0.000E+00	**1.584E+07**
Rb-86	2.348E+05	1.001E-01	**2.348E+05**	2.348E+05	7.827E+04	1.001E-01	**7.827E+04**
Sr-89	6.480E+07	1.915E+07	**8.395E+07**	6.480E+07	2.160E+07	2.285E+07	**4.445E+07**
Sr-90	9.253E+06	4.851E+07	**5.776E+07**	9.253E+06	3.084E+06	5.221E+07	**5.529E+07**
Sr-91	8.105E+07	0.000E+00	**8.105E+07**	8.105E+07	2.702E+07	0.000E+00	**2.702E+07**
Sr-92	8.882E+07	0.000E+00	**8.882E+07**	8.882E+07	2.961E+07	0.000E+00	**2.961E+07**
Y-90	9.615E+06	2.951E+07	**3.913E+07**	9.615E+06	3.205E+06	2.951E+07	**3.272E+07**
Y-91	8.483E+07	3.816E+05	**8.521E+07**	8.483E+07	2.828E+07	3.818E+05	**2.866E+07**
Y-92	8.925E+07	0.000E+00	**8.925E+07**	8.925E+07	2.975E+07	0.000E+00	**2.975E+07**
Y-93	1.046E+08	0.000E+00	**1.046E+08**	1.046E+08	3.487E+07	0.000E+00	**3.487E+07**
Zr-95	1.206E+08	7.839E+05	**1.214E+08**	1.206E+08	4.020E+07	7.839E+05	**4.098E+07**
Zr-97	1.207E+08	0.000E+00	**1.207E+08**	1.207E+08	4.023E+07	0.000E+00	**4.023E+07**
Nb-95	1.220E+08	1.765E+06	**1.238E+08**	1.220E+08	4.067E+07	1.765E+06	**4.243E+07**
Mo-99	1.405E+08	0.000E+00	**1.405E+08**	1.405E+08	4.683E+07	0.000E+00	**4.683E+07**
Tc-99m	1.230E+08	0.000E+00	**1.230E+08**	1.230E+08	4.100E+07	0.000E+00	**4.100E+07**
Ru-103	1.320E+08	7.014E+04	**1.321E+08**	1.320E+08	4.400E+07	7.014E+04	**4.407E+07**
Ru-105	1.010E+08	0.000E+00	**1.010E+08**	1.010E+08	3.367E+07	0.000E+00	**3.367E+07**
Ru-106	6.560E+07	2.211E+07	**8.771E+07**	6.560E+07	2.187E+07	2.211E+07	**4.398E+07**
Rh-105	9.303E+07	0.000E+00	**9.303E+07**	9.303E+07	3.101E+07	0.000E+00	**3.101E+07**
Sb-127	9.609E+06	0.000E+00	**9.609E+06**	9.609E+06	3.203E+06	0.000E+00	**3.203E+06**
Sb-129	2.678E+07	0.000E+00	**2.678E+07**	2.678E+07	8.927E+06	0.000E+00	**8.927E+06**
Te-127	9.546E+06	4.768E+04	**9.594E+06**	9.546E+06	3.182E+06	4.768E+04	**3.230E+06**
Te-127m	1.294E+06	4.868E+04	**1.343E+06**	1.294E+06	4.313E+05	4.868E+04	**4.800E+05**
Te-129	2.637E+07	4.578E+02	**2.637E+07**	2.637E+07	8.790E+06	4.578E+02	**8.790E+06**
Te-129m	3.930E+06	7.032E+02	**3.931E+06**	3.930E+06	1.310E+06	7.032E+02	**1.311E+06**
Te-131m	1.151E+07	0.000E+00	**1.151E+07**	1.151E+07	3.837E+06	0.000E+00	**3.837E+06**
Te-132	1.073E+08	0.000E+00	**1.073E+08**	1.073E+08	3.577E+07	0.000E+00	**3.577E+07**
I-131	7.686E+07	5.587E-07	**7.686E+07**	7.686E+07	2.562E+07	5.587E-07	**2.562E+07**
I-132	1.094E+08	0.000E+00	**1.094E+08**	1.094E+08	3.647E+07	0.000E+00	**3.647E+07**
I-133	1.486E+08	0.000E+00	**1.486E+08**	1.486E+08	4.953E+07	0.000E+00	**4.953E+07**
I-134	1.616E+08	0.000E+00	**1.616E+08**	1.616E+08	5.387E+07	0.000E+00	**5.387E+07**
I-135	1.396E+08	0.000E+00	**1.396E+08**	1.396E+08	4.653E+07	0.000E+00	**4.653E+07**
Xe-133	1.492E+08	6.675E-14	**1.492E+08**	1.492E+08	4.973E+07	6.675E-14	**4.973E+07**
Xe-135	4.333E+07	0.000E+00	**4.333E+07**	4.333E+07	1.444E+07	0.000E+00	**1.444E+07**
Cs-134	2.606E+07	2.171E+07	**4.777E+07**	2.606E+07	8.687E+06	2.172E+07	**3.041E+07**
Cs-136	7.018E+06	9.483E-03	**7.018E+06**	7.018E+06	2.339E+06	9.483E-03	**2.339E+06**
Cs-137	1.284E+07	6.774E+07	**8.058E+07**	1.284E+07	4.280E+06	7.294E+07	**7.722E+07**
Ba-139	1.307E+08	0.000E+00	**1.307E+08**	1.307E+08	4.357E+07	0.000E+00	**4.357E+07**

	Case 2 - Source Term			Case 1 - Source Term			
Nuclide	LOCA Source (Curies)	20 Years (6 2/3 Cores) (Curies)	Total (Curies)	LOCA Source (Curies)	1/3 LOCA Source (Curies)	22 Years (7 2/3 Cores) (Curies)	Total (Curies)
Ba-140	1.258E+08	1.060E-01	**1.258E+08**	1.258E+08	4.193E+07	1.060E-01	**4.193E+07**
La-140	1.310E+08	1.220E-01	**1.310E+08**	1.310E+08	4.367E+07	1.220E-01	**4.367E+07**
La-141	1.190E+08	0.000E+00	**1.190E+08**	1.190E+08	3.967E+07	0.000E+00	**3.967E+07**
La-142	1.146E+08	0.000E+00	**1.146E+08**	1.146E+08	3.820E+07	0.000E+00	**3.820E+07**
Ce-141	1.208E+08	1.680E+04	**1.208E+08**	1.208E+08	4.027E+07	1.680E+04	**4.028E+07**
Ce-143	1.094E+08	0.000E+00	**1.094E+08**	1.094E+08	3.647E+07	0.000E+00	**3.647E+07**
Ce-144	1.014E+08	2.353E+07	**1.249E+08**	1.014E+08	3.380E+07	2.353E+07	**5.733E+07**
Pr-143	1.088E+08	3.168E-01	**1.088E+08**	1.088E+08	3.627E+07	3.168E-01	**3.627E+07**
Nd-147	4.809E+07	1.837E-03	**4.809E+07**	4.809E+07	1.603E+07	1.837E-03	**1.603E+07**
Np-239	1.960E+09	3.258E+04	**1.960E+09**	1.960E+09	6.533E+08	3.583E+04	**6.534E+08**
Pu-238	5.475E+05	3.604E+06	**4.152E+06**	5.475E+05	1.825E+05	3.935E+06	**4.118E+06**
Pu-239	3.828E+04	2.587E+05	**2.970E+05**	3.828E+04	1.276E+04	2.846E+05	**2.974E+05**
Pu-240	6.456E+04	4.359E+05	**5.005E+05**	6.456E+04	2.152E+04	4.800E+05	**5.015E+05**
Pu-241	1.626E+07	6.795E+07	**8.421E+07**	1.626E+07	5.420E+06	7.180E+07	**7.722E+07**
Am-241	2.152E+04	1.473E+06	**1.495E+06**	2.152E+04	7.173E+03	1.715E+06	**1.722E+06**
Cm-242	6.998E+06	6.451E+05	**7.643E+06**	6.998E+06	2.333E+06	6.465E+05	**2.979E+06**
Cm-244	1.053E+06	4.814E+06	**5.867E+06**	1.053E+06	3.510E+05	5.122E+06	**5.473E+06**
I-130	4.626E+06	0.000E+00	**4.626E+06**	4.626E+06	1.542E+06	0.000E+00	**1.542E+06**
Kr-83m	8.634E+06	0.000E+00	**8.634E+06**	8.634E+06	2.878E+06	0.000E+00	**2.878E+06**
Xe-138	1.198E+08	0.000E+00	**1.198E+08**	1.198E+08	3.993E+07	0.000E+00	**3.993E+07**
Xe-131m	8.582E+05	5.145E-04	**8.582E+05**	8.582E+05	2.861E+05	5.145E-04	**2.861E+05**
Xe-133m	4.765E+06	0.000E+00	**4.765E+06**	4.765E+06	1.588E+06	0.000E+00	**1.588E+06**
Xe-135m	3.081E+07	0.000E+00	**3.081E+07**	3.081E+07	1.027E+07	0.000E+00	**1.027E+07**
Cs-138	1.334E+08	0.000E+00	**1.334E+08**	1.334E+08	4.447E+07	0.000E+00	**4.447E+07**
Cs-134m	5.846E+06	0.000E+00	**5.846E+06**	5.846E+06	1.949E+06	0.000E+00	**1.949E+06**
Rb-88	4.841E+07	0.000E+00	**4.841E+07**	4.841E+07	1.614E+07	0.000E+00	**1.614E+07**
Rb-89	6.176E+07	0.000E+00	**6.176E+07**	6.176E+07	2.059E+07	0.000E+00	**2.059E+07**
Sb-124	2.157E+05	1.088E+03	**2.168E+05**	2.157E+05	7.190E+04	1.088E+03	**7.299E+04**
Sb-125	1.797E+06	2.106E+06	**3.903E+06**	1.797E+06	5.990E+05	2.112E+06	**2.711E+06**
Sb-126	1.244E+05	1.019E+02	**1.245E+05**	1.244E+05	4.147E+04	1.121E+02	**4.158E+04**
Te-131	6.773E+07	0.000E+00	**6.773E+07**	6.773E+07	2.258E+07	0.000E+00	**2.258E+07**
Te-133	8.797E+07	0.000E+00	**8.797E+07**	8.797E+07	2.932E+07	0.000E+00	**2.932E+07**
Te-134	1.188E+08	0.000E+00	**1.188E+08**	1.188E+08	3.960E+07	0.000E+00	**3.960E+07**
Te-125m	3.947E+05	5.137E+05	**9.084E+05**	3.947E+05	1.316E+05	5.151E+05	**6.467E+05**
Te-133m	5.267E+07	0.000E+00	**5.267E+07**	5.267E+07	1.756E+07	0.000E+00	**1.756E+07**
Ba-141	1.184E+08	0.000E+00	**1.184E+08**	1.184E+08	3.947E+07	0.000E+00	**3.947E+07**
Ba-137m	1.216E+07	6.408E+07	**7.624E+07**	1.216E+07	4.053E+06	6.900E+07	**7.305E+07**
Pd-109	3.771E+07	0.000E+00	**3.771E+07**	3.771E+07	1.257E+07	0.000E+00	**1.257E+07**
Rh-106	7.109E+07	2.211E+07	**9.320E+07**	7.109E+07	2.370E+07	2.211E+07	**4.581E+07**
Rh-103m	1.189E+08	6.323E+04	**1.190E+08**	1.189E+08	3.963E+07	6.323E+04	**3.970E+07**
Tc-101	1.293E+08	0.000E+00	**1.293E+08**	1.293E+08	4.310E+07	0.000E+00	**4.310E+07**
Eu-154	1.606E+06	5.104E+06	**6.710E+06**	1.606E+06	5.353E+05	5.294E+06	**5.829E+06**
Eu-155	1.088E+06	2.269E+06	**3.357E+06**	1.088E+06	3.627E+05	2.305E+06	**2.668E+06**
Eu-156	2.847E+07	5.466E-01	**2.847E+07**	2.847E+07	9.490E+06	5.466E-01	**9.490E+06**
La-143	1.086E+08	0.000E+00	**1.086E+08**	1.086E+08	3.620E+07	0.000E+00	**3.620E+07**
Nb-97	1.218E+08	0.000E+00	**1.218E+08**	1.218E+08	4.060E+07	0.000E+00	**4.060E+07**

	Case 2 - Source Term			Case 1 - Source Term			
Nuclide	**LOCA Source (Curies)**	**20 Years (6 2/3 Cores) (Curies)**	**Total (Curies)**	**LOCA Source (Curies)**	**1/3 LOCA Source (Curies)**	**22 Years (7 2/3 Cores) (Curies)**	**Total (Curies)**
Nb-95m	8.606E+05	5.816E+03	**8.664E+05**	8.606E+05	2.869E+05	5.816E+03	**2.927E+05**
Pm-147	1.187E+07	1.364E+07	**2.551E+07**	1.187E+07	3.957E+06	1.367E+07	**1.763E+07**
Pm-148	2.220E+07	1.166E+02	**2.220E+07**	2.220E+07	7.400E+06	1.166E+02	**7.400E+06**
Pm-149	4.726E+07	0.000E+00	**4.726E+07**	4.726E+07	1.575E+07	0.000E+00	**1.575E+07**
Pm-151	1.686E+07	0.000E+00	**1.686E+07**	1.686E+07	5.620E+06	0.000E+00	**5.620E+06**
Pm-148m	2.843E+06	2.071E+03	**2.845E+06**	2.843E+06	9.477E+05	2.071E+03	**9.497E+05**
Pr-144	1.021E+08	2.354E+07	**1.256E+08**	1.021E+08	3.403E+07	2.354E+07	**5.757E+07**
Pr-144m	1.218E+06	2.824E+05	**1.500E+06**	1.218E+06	4.060E+05	2.824E+05	**6.884E+05**
Sm-153	5.086E+07	0.000E+00	**5.086E+07**	5.086E+07	1.695E+07	0.000E+00	**1.695E+07**
Y-94	1.062E+08	0.000E+00	**1.062E+08**	1.062E+08	3.540E+07	0.000E+00	**3.540E+07**
Y-95	1.152E+08	0.000E+00	**1.152E+08**	1.152E+08	3.840E+07	0.000E+00	**3.840E+07**
Y-91m	4.705E+07	0.000E+00	**4.705E+07**	4.705E+07	1.568E+07	0.000E+00	**1.568E+07**
Br-82	6.291E+05	0.000E+00	**6.291E+05**	6.291E+05	2.097E+05	0.000E+00	**2.097E+05**
Br-83	8.606E+06	0.000E+00	**8.606E+06**	8.606E+06	2.869E+06	0.000E+00	**2.869E+06**
Br-84	1.470E+07	0.000E+00	**1.470E+07**	1.470E+07	4.900E+06	0.000E+00	**4.900E+06**
Am-242	1.041E+07	1.779E+04	**1.043E+07**	1.041E+07	3.470E+06	1.949E+04	**3.489E+06**
Np-238	5.399E+07	8.943E+01	**5.399E+07**	5.399E+07	1.800E+07	9.793E+01	**1.800E+07**
Pu-243	6.043E+07	1.144E-03	**6.043E+07**	6.043E+07	2.014E+07	1.259E-03	**2.014E+07**

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
Stack/Plant Vent	N CR intake	184	56.1	59.75	18.2	48.08	14.6	58
Stack/Plant Vent	S CR intake	184	56.1	59.75	18.2	126.69	38.6	354
RWT	N CR intake	48.22	14.6	59.75	18.2	245.31	74.7	65
RWT	S CR intake	48.22	14.6	59.75	18.2	263.64	80.3	39
FHB Closest Point	N CR intake	43.25	13.2	59.75	18.2	120.6	36.7	48
FHB Closest Point	S CR intake	43.25	13.2	59.75	18.2	184.26	56.1	11
Aux. Bldg. Louver L-7B	N CR intake	38.17	11.6	59.75	18.2	123.77	37.7	72
Aux. Bldg. Louver L-7A	S CR intake	38.17	11.6	59.75	18.2	136.97	41.7	34
Condenser	N CR intake	5.25	1.6	59.75	18.2	153.24	46.7	245
Closest MSSV/ADV	N CR intake	48	14.6	59.75	18.2	108.59	33.0	300
Closest MSSV/ADV	S CR intake	48	14.6	59.75	18.2	215.01	65.5	316
Closest Feedwater Line Point	N CR intake	17	5.2	59.75	18.2	83.29	25.3	306
FHB Closest Point	Midpoint between intakes	43.25	13.2	59.75	18.2	142.19	43.3	25
Stack/Plant Vent	Midpoint between intakes	184	56.1	59.75	18.2	74.85	22.8	8
RWT	Midpoint between intakes	48.22	14.6	59.75	18.2	244.91	74.6	52
Aux. Bldg. Louver L-7A	Midpoint between intakes	38.17	11.6	59.75	18.2	118.59	36.1	59
Closest MSSV/ADV	Midpoint between intakes	48	14.6	59.75	18.2	161.58	49.2	310

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
Closest Feedwater Line Point	Midpoint between intakes	17	5.2	59.75	18.2	138.15	42.1	315

Notes:

1. Release heights are calculated as 19 feet less than the reference elevations to account for the plant grade elevation.

2. The FHB closest point release elevation is taken as the roof elevation since the SW corner of the roof is the closest building point to the intakes.

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

4. The only release/receptor combination that does not have the intakes in the same wind direction window from the release point is for the releases from the plant stack. All other release points analyzed result in both control room intakes being in the same wind direction window. Therefore, credit may be taken for intake dilution only for releases from the plant stack.

5. The receptor point for the "midpoint between intakes" is taken as being on the outside of the control room (and H&V room) east wall. The receptor elevation is taken as the average of the receptor elevations for the two outside air intakes.

6. Atmospheric dispersion factors for the releases to the midpoint between the control room intakes are required for the limiting cases to be used during the time period when the control room intakes are isolated. This midpoint receptor location is used to calculate the *X/Q* value to be used for the unfiltered control room inleakage dose.

7. The closest containment/shield building penetration to the intakes that is directly exposed to the atmosphere is the closest feedwater line penetration.

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 unfavorable intake prior to intake isolation, the midpoint between the intakes for during isolation, as well as values for the favorable intake due to the manual selection of the favorable control room intake after unisolation and initiation of filtered air make-up. These values are not corrected for Control Room Occupancy Factors but do include taking credit for dilution where allowed. Based on the layout of the site, the only cases that may take credit for dilution are when the releases are from the plant vent stack. However, dilution is not credited during the time period when the control room intakes are isolated for these cases.

* Indicates credit for dilution taken for this case.

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 *	Stack/Plant Vent *	N CR intake*	2.35E-03				
B *	Stack/Plant Vent *	S CR intake*	6.68E-04	4.55E-04	2.11E-04	1.26E-04	9.25E-05
C	RWT	N CR intake	1.38E-03				
D	RWT	S CR intake	1.10E-03	9.30E-04	3.96E-04	2.94E-04	2.28E-04
E	FHB Closest Point	N CR intake	4.93E-03				
F	FHB Closest Point	S CR intake	2.00E-03	1.40E-03	6.36E-04	4.22E-04	3.09E-04
G	Aux. Bldg. Louver L-7B	N CR intake	4.85E-03				
H	Aux. Bldg. Louver L-7A	S CR intake	3.59E-03	2.94E-03	1.24E-03	8.84E-04	6.91E-04
I	Condenser	N CR intake	2.47E-03				
J	Closest MSSV/ADV	N CR intake	5.87E-03				
K	Closest MSSV/ADV	S CR intake	1.58E-03	1.23E-03	4.90E-04	3.59E-04	2.69E-04
L	Closest Feedwater Line Point	N CR intake	7.30E-03				
M	FHB Closest Point	Midpoint between intakes	3.26E-03				
N	Stack/Plant Vent	Midpoint between intakes	3.78E-03				
O	RWT	Midpoint between intakes	1.33E-03				

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*
P	Aux. Bldg. Louver L-7A	Midpoint between intakes	5.04E-03				
Q	Closest MSSV/ ADV	Midpoint between intakes	2.74E-03				
R	Closest Feedwater Line Point	Midpoint between intakes	3.17E-03				

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

Event	Prior to CR Isolation	During CR Isolation	After Initiation of Filtered Air Make-up
LOCA:			
- SBVS Leakage	L	R (prior to SBVS drawdown) N (after SBVS drawdown)	B
- SBVS Bypass Leakage	A	N	B
- ECCS Leakage	G	P	H
- RWT Backleakage	C	O	D
- Cont. Purge/H_2 Purge	A	N	B
FHA	E	M	F
MSLB:			
- Outside Cont.	J	Q	K
- Inside Cont.	L	R (prior to SBVS drawdown) N (after SBVS drawdown)	B
SGTR	I (release to Condenser prior to LOOP/Turb. Trip) J (after Turbine Trip)	Q	K
Locked Rotor	J	Q	K
CEA Ejection:			
- Primary Leakage	L	R (prior to SBVS drawdown) N (after SBVS drawdown)	B
- Secondary Side Release	J	Q	K
IOMSSV	J	Q	K
Cask Drop	E	M	F
WGDT Rupture	A	N	B

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	1.03E-04	9.97E-05
0-8 hours	5.69E-05	5.47E-05
8-24 hours	4.22E-05	4.05E-05
1-4 days	2.22E-05	2.11E-05
4-30 days	8.79E-06	8.29E-06

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	2754 MW_{th} (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 4.5 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	480,538 lb_m
Core Fission Product Inventory	Table 1.7.4-1
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.5% (by weight)/day 0.25% (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 15 minutes to 30 days)</u> Sump Volume (minimum) ECCS Leakage to RAB (2 times allowed value) Flashing Fraction (all elemental Iodine assumed to be released) Chemical form of the iodine in the sump water (based on pH history) 0 to 1 hour after 1 hour Release ECCS Area Filtration Efficiency	 57,000 ft^3 4510 cc/hr 10% assumed 95 % aerosol, 4.85 elemental, and 0.15% organic 99.6% aerosol, 0.25% elemental, and 0.15% organic Particulate – 99% Elemental – 97.5% Organic – 97.5%

Input/Assumption	Value
RWT Back-leakage Sump Volume (at time of recirculation) RWT Piping Volume ECCS Leakage to RWT (2 times allowed value) Flashing Fraction (elemental Iodine assumed to be released into tank space based upon partition factor) RWT liquid/vapor Elemental Iodine partition factor Chemical form of iodine in the RWT (based on Sump and RWT pH history the model will add 2.5% of regenerated elemental iodine to the assumed sump concentration) 0 to 1 hour after 1 hour	 60,958 ft^3 500 ft^3 2 gpm 0 % assumed 25 92.5% aerosol, 7.35% elemental, and 0.15% organic 97.1% aerosol, 2.75% elemental, and 0.15% organic
Initial RWT Liquid Inventory (minimum)	51,415 gallons
Release from RWT Vapor Space	Table 2.1-3
Containment Purge Release	42,000 cfm for 30 seconds
Removal Inputs:	
Containment Particulate/Aerosol Natural Deposition (only credited in unsprayed regions)	0.1/hour
Containment Elemental Iodine Natural/Wall Deposition	2.89/hour
Containment Spray Region Volume	2,155,160 ft^3
Containment Unsprayed Region Volume	350,840 ft^3
Flowrate between sprayed and unsprayed volumes	11,695 cfm
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Particulate Iodine Time to reach DF of 50	 20/hour 2.065 hours 5.53/hour 2.507 hours
Spray Initiation Time	0.01806 hours
Control Room Ventilation System Time of automatic control room isolation Time of manual control room unisolation	Table 1.6.3-1 50 seconds 1.5 hrs

Input/Assumption	Value
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 97.5% Organic – 97.5%
Secondary Containment Drawdown Time	120 seconds
Secondary Containment Bypass Fraction	9.6%
Containment Purge Filtration	99% particulate, elemental iodine and organic iodine
Transport Inputs:	
Containment Release Secondary Containment release prior to drawdown	Nearest Containment penetration to CR ventilation intake
Containment Release Secondary Containment release after drawdown	Plant stack
Containment Release Secondary Containment Bypass Leakage	Plant stack
ECCS Leakage	ECCS exhaust louver
RWT Backleakage	RWT
Containment Purge	Plant Stack
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	RG 1.183 Sections 4.1.3 and 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 Release from RWT

Time (hours)	Adjusted Release Rate* (cfm)
0	0.0468
2	0.0468
8	0.0467
24	0.0466
48	0.0464
96	0.0459
150	0.0454
250	0.0446
350	0.0437
450	0.0428
550	0.0419
650	0.0410
720	0.0404

* Air/vapor released from RWT (due to displacement by incoming leakage and expansion due to diurnal heating) divided by the partition coefficient (the partition coefficient equals 25).

Table 2.1-4 LOCA Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	1.20	2.56	4.88
Acceptance Criteria	25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

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

Input/Assumption	Value
Core Power Level Before Shutdown	2754 MW_{th} (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Discharged Fuel Assembly Burnup	45,000 – 62,000 MWD/MTU
Fuel Enrichment	3.0 – 4.5 w/o
Maximum Radial Peaking Factor	1.7
Number of Fuel Assemblies in the Core	217
Number of Fuel Assemblies Damaged	1
Delay Before Spent Fuel Movement	72 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 – 500 Organic – 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Chemical Form of Iodine Above Pool	Elemental – 57% Organic – 43%
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	50 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Time of Control Room Filtered Makeup Flow	1.5 hours
Breathing Rates	RG 1.183 Sections 4.1.3 and 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)
Elemental Iodine DF = 500			
Bounding FHA (containment or FHB)	0.25	0.25	2.89
Elemental Iodine DF = 285			
Bounding FHA (containment or FHB)	0.33	0.32	3.72
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

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

Input/Assumption	Value
Core Power Level	2754 MW_{th} (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 4.5 w/o
Maximum Radial Peaking Factor	1.7
% DNB Fuel for MSLB Outside of Containment	3.5%
% DNB Fuel for MSLB Inside of Containment	100%
% Fuel Centerline Melt for MSLB Outside of Containment	0.81%
% Fuel Centerline Melt for MSLB Inside of Containment	16%
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	RG 1.183, Sections 3.1, 3.2, and Table 3
Release From Fuel Centerline Melt Fuel	RG 1.183, Sections 3.1, 3.2, and Table 3 and Section 1 of Appendix H to RG 1.183
Steam Generator Secondary Side Partition Coefficient	Unaffected SG - 100 Faulted SG - None
Steam Generator Tube Leakage	0.15 gpm per SG
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	12 hours
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.506E+06 ft^3 0.5% (by weight)/day 0.25% (by weight)/day
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 97.5% Organic – 97.5%
Secondary Containment Drawdown Time	0.5 hours (conservative assumption)
Secondary Containment Bypass Fraction	9.6%
RCS Mass	minimum – 411,500 lb_m maximum – 480,538 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 – 105,000 lb_m (one SG) maximum – 205,000 lb_m (one SG) Maximum mass used for faulted SG to maximize secondary side dose contribution. Minimum mass used for intact SG to maximize steam release nuclide concentration.
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97 % Organic – 3%

Input/Assumption	**Value**
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	50 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Time of Control Room Filtered Makeup Flow	1.5 hours
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 2.89 hr^{-1} Organic Iodine – None

Table 2.3-2 MSLB Steam Release Rate

Time (hours)	Intact SG Steam Release Rate (lb_m/min)
0 – 0.25	7902
0.25 – 0.50	4196
0.50 – 0.75	4705
0.75 – 1.0	5362
1.0 – 2.0	4901
2.0 – 4.0	3525
4.0 – 6.0	3245
6.0 – 8.0	2977
8.0 – 720.0	0.0

Table 2.3-3 MSLB Dose Consequences

Case	Fuel Failure	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
MSLB – Outside of Containment	3.5% DNB	0.31	0.85	4.87
MSLB – Outside of Containment	0.81% FCM	0.35	0.89	4.86
MSLB – Inside of Containment	100% DNB	0.32	0.80	3.86
MSLB – Inside of Containment	16% FCM	0.59	1.29	4.75
Acceptance Criteria		25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

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

Input/Assumption	Value
Core Power Level	2754 MW_{th} (2700 + 2%)
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	60μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	Faulted SG - 0.15 gpm Intact SG – 0.15 gpm
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	12 hours
RCS Mass	Pre-accident spike – 480,472 lb_m Concurrent spike – 438,783 lb_m
SG Secondary Side Mass	minimum – 105,000 lb_m (one SG)
Integrated Mass Release	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 SG – 100
Break Flow Flash Fraction	8.76%
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 Time of manual control room unisolation	Table 1.6.3-1 385 seconds 1.5 hours
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 Integrated Mass Releases [1]

Time (hours)	Break Flow in Ruptured SG (lb_m)	Steam Release from Ruptured SG (lb_m)	Steam Release from Unaffected SG (lb_m)
0 – 0.5	86,814	0 – 0.0915 hrs : 546,210 (via Condenser) 0.0915 – 0.5 hrs: 85,211 (via MSSV)	543,030 (via Condenser) 83,258 (via MSSVs)
0.5 – 2.0	0	0	580,606 (via ADVs)
2 – 8	N/A	N/A	921,445

[1] Flowrate assumed to be constant within time period

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

Isotope	Activity (μCi/gm)
Iodine-131	47.52
Iodine-132	13.05
Iodine-133	67.74
Iodine-134	7.422
Iodine-135	32.32

Table 2.4-4 Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	128 gpm
Assumed Letdown Flow *	150 gpm at 120°F, 2250 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	438,783 lb_m

* maximum letdown flow plus uncertainty

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

Isotope	Activity Appearance Rate (Ci/min)	Total 8 hour production (Ci)
Iodine-131	1.616E+02	7.756E+04
Iodine-132	1.163E+02	5.585E+04
Iodine-133	2.676E+02	1.285E+05
Iodine-134	1.334E+02	6.405E+04
Iodine-135	1.705E+02	8.186E+04

Table 2.4-6 SGTR Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
SGTR pre-accident iodine spike	0.29	0.28	3.31
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	0.10	0.10	1.10
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

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

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

Input/Assumption	Value
Core Power Level	2754 MW_{th} (2700 + 2%)
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	3.0 – 4.5w/o
Maximum Radial Peaking Factor	1.7
Fuel Failure	13.7%
RCS Mass	minimum – 411,500 lb_m maximum – 480,538 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	0.3 gpm total (0.15 gpm per SG)
Time to establish shutdown cooling and terminate release	8 hours
SG Minimum Mass (per SG)	105,000 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 Time of manual control room unisolation	Table 1.6.3-1 50 seconds 1.5 hrs
Breathing Rates Offsite Onsite	 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 – 0.25	7902
0.25 – 0.50	4196
0.50 – 0.75	4705
0.75 – 1.0	5362
1.0 – 2.0	4901
2.0 – 4.0	3525
4.0 – 6.0	3245
6.0 – 8.0	2977
8.0 – 720.0	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	0.10	0.23	1.64
Acceptance Criteria	2.5	2.5	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 2.6-1
Control Element Assembly (CEA) Ejection – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2754 MW_{th} (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 4.5 w/o
Maximum Radial Peaking Factor	1.7
% DNB Fuel	9.5%
% Fuel Centerline Melt	0.5%
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	0.3 gpm total
Time to establish shutdown cooling and terminate steam release	8 hours
RCS Mass	minimum – 411,500 lb_m Minimum mass used for fuel failure dose contribution to maximum SG tube leakage activity.
SG Secondary Side Mass	minimum – 105,000 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	50 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Time of Control Room Filtered Makeup Flow	1.5 hours
Breathing Rates	RG 1.183 Sections 4.1.3 and 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	2.506E+06 ft^3 0.5% (by weight)/day 0.25% (by weight)/day
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 97.5% Organic – 97.5%
Secondary Containment Drawdown Time	4 minutes (240 seconds)
Secondary Containment Bypass Fraction	9.6%
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 2.89 hr^{-1} Organic Iodine – None

Table 2.6-2 CEA Ejection Steam Release Rate

Time (hours)	SG Steam Release Rate (lb_m/min)
0 – 0.25	7902
0.25 – 0.50	4196
0.50 – 0.75	4705
0.75 – 1.0	5362
1.0 – 2.0	4901
2.0 – 4.0	3525
4.0 – 6.0	3245
6.0 – 8.0	2977
8.0 – 720.0	0.0

Table 2.6-3 CEA Ejection Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
CEA Ejection – Containment Release	0.13	0.28	1.67
CEA Ejection – Secondary Release	0.23	0.47	3.03
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 2.7-1
IOMSSV – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2754 MW_{th} (2700 + 2%)
RCS Activity prior to accident	Table 1.7.2-1
RCS Mass	480,538 lb_m (Maximum Mass used for RCS activity)
Primary-to-Secondary Leakage Rate	0.3 gpm total (0.15 gpm per SG)
Time to establish 212°F in RCS and terminate release	12 hours
SG maximum mass (per SG)	205,000 lb_m
Secondary Side Iodine Activity prior to accident	Table 1.7.3-1
Secondary Side Mass Releases to environment	Table 2.5-2
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 isolation Time of manual control room unisolation	Table 1.6.3-1 50 seconds 1.5 hours
Breathing Rates: Offsite Onsite	 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 2.7-2 IOMSSV Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Inadvertent Opening of a MSSV	0.02	0.02	0.40
Acceptance Criteria	2.5	2.5	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 2.8-1
Spent Fuel Cask Drop– Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	2754 MW_{th} (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Discharged Fuel Assembly Burnup	45,000 – 62,000 MWD/MTU
Fuel Enrichment	3.0 – 4.5 w/o
Number of Fuel Assemblies Damaged	7 2/3 cores
Delay Before Spent Fuel Movement	Case 1 – 1180 hours Case 2 – 1490 hours
Source Terms	Table 1.7.6-1
Water Level Above Damaged Fuel Assembly	23 feet minimum
Iodine Decontamination Factors	Elemental – 500 Organic – 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Chemical Form of Iodine Above Pool	Elemental – 57% Organic – 43%
Atmospheric Dispersion Factors	Table 1.8.1-2 and 1.8.1-3
Time of Control Room Ventilation System Isolation	50 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Time of Control Room Filtered Makeup Flow	1.5 hours
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.8-2 Spent Fuel Cask Drop Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Elemental iodine DF = 500			
Cask Drop - Case 1	3.3351E-04	0.15	1.76
Cask Drop - Case 2	2.4743E-04	0.14	1.74
Elemental iodine DF = 285			
Cask Drop - Case 1	4.5370E-04	0.20	2.42
Cask Drop - Case 2	3.3549E-04	0.20	2.39
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 2.9-1
Waste Gas Decay Tank (WGDT) Rupture – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2700 MW_{th}
WGDT inventory	Table 2.9-2
Tank volume (arbitrary)	144 ft^3
Tank leak rate (arbitrarily high)	1000 cfm
Control Room Ventilation System Time of automatic control room isolation Time of manual control room unisolation	Table 1.6.3-1 50 seconds 1.5 hrs
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 WGDT Source Term (1)

Isotope	Tank Inventory (Curies)
I-131	2.98E-01
I-132	7.70E-04
I-133	4.22E-02
I-134	1.74E-04
I-135	6.71E-03
Kr-85m	3.97E+01
Kr-85	6.46E+04
Kr-87	7.95E+00
Kr-88	5.96E+01
Xe-131m	8.94E+03
Xe-133	2.73E+05
Xe-135	4.22E+02
Xe-138	8.45E-01

(1) 24.84 times the values from St. Lucie Unit 2 UFSAR Table 15.7.4.1-2

Table 2.9-3 WGDT Rupture Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
WGDT	0.17	0.17	0.50
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 3-1

St. Lucie Plant, Unit No. 1
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	580	1.20	2.56	4.88
MSLB – Outside of Containment (3.5% DNB)	580	0.31	0.85	4.87
MSLB – Outside of Containment (0.81% FCM)	580	0.35	0.89	4.86
MSLB – Inside of Containment (100% DNB)	580	0.32	0.80	3.86
MSLB – Inside of Containment (16% FCM)	580	0.59	1.29	4.75
SGTR Pre-accident Iodine Spike	1000	0.29	0.28	3.31
Acceptance Criteria		≤ 25(3)	≤ 25(3)	≤ 5(4)
SGTR Concurrent Iodine Spike	1000	0.10	0.10	1.10
Locked Rotor (13.7 % DNB)	1000	0.10	0.23	1.64
IOMSSV *	1000	0.02	0.02	0.40
Acceptance Criteria		≤ 2.5 (3)	≤ 2.5 (3)	≤ 5 (4)
Bounding FHA (in containment or FHB, Elemental iodine DF = 285)	1000	0.33	0.32	3.72
CEA Ejection – Containment Release (9.5 % DNB, 0.5 % FCM)	1000	0.13	0.28	1.67
CEA Ejection – Secondary Side Release (9.5 % DNB, 0.5 % FCM)	1000	0.23	0.47	3.03
Spent Fuel Cask Drop – Case 1* (Elemental iodine DF=285)	1000	4.5370E-04	0.20	2.42
Spent Fuel Cask Drop – Case 2* (Elemental iodine DF=285)	1000	3.3549E-04	0.20	2.39
WGDT	1000	0.17	0.17	0.50
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 appropriate event summary in Section 2.0 for basis of acceptance criteria