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

FPL

July 16, 2007

L-2007-087
10 CFR 50.90

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

RE: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment
Alternative Source Term and Conforming Amendment

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License NPF-16 for St. Lucie Unit 2. FPL proposes to revise the St. Lucie Unit 2 licensing bases to adopt the alternative source term (AST) as allowed in 10 CFR 50.67.

Attachment 1 is a description of the proposed changes and the supporting justification including the Determination of No Significant Hazards and Environmental Considerations. Attachment 2 provides marked up copies of the proposed Technical Specification changes. Attachment 3 provides copies of the word processed TS pages. Attachment 4 provides information only copies of the marked up TS Bases pages. Enclosure 1 is Numerical Applications, Inc., NAI-1101-044, "AST Licensing Technical Report for St. Lucie Unit 2," Revision 2.

The proposed amendment has been reviewed in accordance with the FPL QATR requirements. 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.

FPL submitted proposed license amendments via FPL letter L-2007-084 to implement TSTF-448, Rev. 3. This AST license amendment request needs to be approved by the NRC before or at the same time as the license amendment request for TSTF-448, Rev. 3. Therefore, FPL requests that the proposed license amendment be issued concurrent with the TSTF-448, Rev. 3. license amendments, with the amendments being implemented within 90 days.

Please contact Ken Frehafer at (772) 467-7748 if there are any questions about this submittal.

A001

NRR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 16th day of July 2007.

Sincerely,

Gordon Johnston

Gordon L. Johnston
Site Vice President
St. Lucie Plant

GLJ/KWF

Attachments

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

Regulatory Assessment of the Proposed Implementation of the
Alternative Radiological Source Term Methodology
for the St. Lucie Plant, Unit No. 2

Introduction

The current St. Lucie Plant, Unit No. 2, 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 with the exception of the Steam Generator Tube Rupture event which was revised to the Alternative Source Term methodology as part of License Amendment 138.

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 terms (ASTs). Part 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of the affected design basis accidents. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183.

As documented in 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 No. 2 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
- Letdown Line Break
- Feedwater Line Break (FWLB)

Each accident and the specific input assumptions are described in the Numerical Applications, Inc. (NAI) "AST Licensing Technical Report for St. Lucie Unit 2," NAI-1101-044, Revision 2 (Attachment 3). The inputs and assumptions related to the steam generators are based on the replacement steam generators which are scheduled for installation during the fall 2007 refueling

outage. These analyses provide for a bounding allowable control room unfiltered air inleakage of 435 cfm. The use of 435 cfm as a design basis value is expected to be 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.

Description of Proposed Amendment

Florida Power and Light (FPL) Company proposes to revise the St. Lucie Plant, Unit No. 2, licensing basis to implement the AST as allowed by 10 CFR 50.67 and 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.
- A steam generator tube leakage rate (for accident analysis) that exceeds the revised Technical Specification operational leakage limit submitted for approval as part of the Steam Generator Tube Integrity License Amendment Request (Ref. 13) is utilized.
- Credit for the ECCS Area Ventilation System HEPA filters and charcoal adsorbers is being taken.
- An SBVS bypass leakage value selected that is more restrictive than the current Technical Specification limit is utilized.

Accordingly, the following changes to the St. Lucie, Unit No. 2, 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," as the source of thyroid dose conversion factors.

- Surveillance Requirement 4.6.6.1 is revised to relocate the HEPA filter, charcoal adsorber, flow rate and heater surveillance test acceptance criteria for the Shield Building Ventilation System to the Ventilation Filter Testing Program in TS Section 6.8.4.k.

- Surveillance Requirement 4.7.8 is revised to relocate the flow rate surveillance test acceptance criterion for the ECCS Area Ventilation System to the Ventilation Filter Testing Program in TS Section 6.8.4.k.

- 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 12% to 9.6%.

- The HEPA filter and charcoal adsorber test acceptance criteria for the Shield Building Ventilation System relocated to the VFTP are revised as follows:
  - The HEPA filter efficiency test criterion is increased from 99.825% to 99.95%
  - The inplace charcoal adsorber efficiency test acceptance criterion is increased from 99% to 99.95%
  - The laboratory charcoal adsorber efficiency test acceptance criterion is increased from 90% to 97.5%

- The following ECCS Area Ventilation System HEPA filter and charcoal adsorber efficiency test criteria are added to the VFTP:
  - HEPA filter 99.95%
  - Inplace charcoal adsorber 99.95%
  - Laboratory charcoal adsorber 97.5%

- ECCS Area Ventilation System pressure drop test criteria are added to the VFTP.

- In the VFTP (TS 6.8.4.k), reference to Regulatory Guide 1.52, Revision 2 is replaced with reference to Regulatory Guide 1.52, Revision 3.

- In the Surveillance Requirements for the Shield Building Ventilation System (SR 4.6.6.1), as well as in the VFTP (TS 6.8.4.k), reference to ANSI N-510-1975 is replaced with reference to ASME N510-1989.

- The accident induced leakage performance criteria of the Steam Generator (SG) Program (TS Section 6.8.4.l) submitted for approval as part of the Steam Generator Tube Integrity License Amendment Request (Ref. 13) is changed to 0.5 gpm total through all SGs and 0.25 gpm through any one SG.

Justification of Proposed Technical Specification Changes

- Revision of the definition of Dose Equivalent I-131 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," as the source of thyroid dose conversion factors is consistent with the guidance provided in RG 1.183. In the dose calculations, the dose conversion factors referenced in the definition of Dose Equivalent I-131 are used to adjust the initial primary coolant iodine activities for use in the dose calculations. Use of thyroid dose conversion factors (versus effective dose conversion

factors for inhalation or CEDE doses) results in slightly more conservative total iodine concentrations in the primary coolant and, therefore, slightly higher doses.

- Relocation of the HEPA filter, charcoal adsorber, flow rate and heater surveillance test acceptance criteria for the Shield Building Ventilation System, as well as relocation of the flow rate surveillance test criteria for the ECCS Area Ventilation System to the Ventilation Filter Testing Program (VFTP) provides consistency with the existing format for the Control Room Emergency Ventilation System filter testing requirements which is modeled after the format of the VFTP in NUREG-1432, Standard Technical Specifications Combustion Engineering Plants.

  All of the current ventilation filter testing requirements are included in the proposed VFTP. As revised, the filter train operational-type surveillance tests remain in the LCO/Surveillance section, while the direction for the post maintenance or preventative maintenance tests are stated to be in accordance with the VFTP. The VFTP includes all applicable TS surveillance limits.

  The testing methodology requirements are met by requiring that the tests be performed in accordance with ASME N510-1989 and ASTM D3803-1989, as applicable. The frequency requirements are met by describing the VFTP as a program that tests "at the frequencies specified in Regulatory Guide 1.52, Revision 3." These testing frequencies specify off-normal as well as normal (i.e., "scheduled") testing and align with the current TS testing frequencies. Off-normal testing requirements include HEPA and charcoal adsorber leak testing following filter or cell replacements, following filter train contact with foreign fumes and following train maintenance activities.

  The proposed change to delete SRs 4.6.6.1.b.5 and 4.7.8.b.1 is an administrative deletion only as these requirements are delineated in the RG 1.52 inplace HEPA and charcoal adsorber testing requirements.

- Reduction of the acceptance criterion for secondary containment bypass leakage paths (i.e. Shield Building Bypass Leakage) from 12% to 9.6% improves (i.e., increases) the allowable control room unfiltered inleakage. The 9.6% value is supported by plant leakage test results.

- The revised test acceptance criteria for the HEPA filters and the charcoal adsorbers for the Shield Building Ventilation Systems, as well as the addition of ECCS Area Ventilation System HEPA filter and charcoal adsorber efficiency and pressure drop test criteria ensure that the filters and adsorbers meet the filtration efficiencies assumed in the Attachment 3 accident analyses. As described in RG 1.52, Revision 3, the efficiency assumptions allowed are dependent on the test acceptance criteria. The revised test acceptance criteria are consistent with the criteria provided in RG 1.52, Revision 3 and support the assumptions of the Attachment 3 accident analyses. The ECCS Area Ventilation System pressure drop test criterion is based on UFSAR Table 9.4-7.

- Replacing reference to Regulatory Guide 1.52, Revision 2 with reference to Regulatory Guide 1.52, Revision 3 reflects the adoption of the requirements of the most current revision of RG 1.52. As described above, the VFTP testing requirements are consistent with the requirements of Revision 3 of RG 1.52 and support the assumptions of the Attachment 3 accident analyses.

- Replacing reference to ANSI N-510-1975 with reference to ASME N510-1989 is consistent with RG 1.52, Revision 3. Revision 3 of RG 1.52 states that ESF atmosphere cleanup systems tested to ASME N510-1989 (or its earlier versions) are considered adequate to protect public health and safety.

- Changing the accident induced leakage performance criterion of the Steam Generator (SG) Program to 0.5 gpm total through all SGs and 0.25 gpm through any one SG, continues to maintain margin to the operational leakage limit specified in the Technical Specifications. A License Amendment Request (Ref. 13) has been submitted which incorporates TSTF-449, Steam Generator Tube Integrity, and changes the Technical Specification operational primary-to-secondary leakage limit to 150 gpd per SG which is roughly equivalent to 0.1 gpm. The limit of 0.25 gpm per SG was chosen to provide additional margin above the 0.1 gpm TS limit. The limit of 0.5 gpm total through all SGs is consistent with the limit of 0.25 gpm per SG and reflects the maximum total allowable leakage.

Accident Source Term

The full core isotopic inventory for St. Lucie Unit 2 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 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 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 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 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 No. 11 for iodine isotopes). The remaining (non-iodine) isotopes are adjusted to achieve the Tech. Spec. 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 Tech. Specs. 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 2 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 2 (NAI-1101-044) 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 No. 2 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 No. 2 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 accident mitigation features. 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 per the St. Lucie Unit 2 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 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, Letdown Line Break, and Feedwater Line Break (FWLB) are made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the St. Lucie Unit No. 2 licensing basis and the calculated doses do not exceed the defined acceptance criteria.

Results of the St. Lucie Unit 2 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 425 cfm. NAI-1101-044, "AST Licensing Technical Report for St. Lucie Unit 2," explains these results and acceptance criteria in more detail.

No Significant Hazards Determination

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazard if 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 the margin of safety. FPL has reviewed this proposed license amendment for FPL's St. Lucie Unit No. 2 and determined that its adaptation would not involve a significant hazards determination. The bases for this determination are:

This proposed 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 No. 2 which demonstrate that the dose consequences remain below limits specified in NRC Regulatory

Guide 1.183 and 10 CFR 50.67. The proposed changes do not modify the design or operation of the plant. The use of the AST only changes 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 are consistent with, or more restrictive than, the current TS requirements. None of the affected systems, components or programs are related to accident initiators.

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. Neither implementation of the alternative source term methodology nor establishing more restrictive TS requirements have the capability to introduce any new failure mechanisms or cause any analyzed accident to progress in a different manner.

   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 proposed Technical Specification changes are consistent with, or more restrictive than, the current TS requirements. These TS requirements support the AST revisions to the limiting design basis accidents. As such, the current plant margin of safety is preserved. 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 Considerations

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 1.22(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:

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. The occupational dose received from performing the proposed surveillance for the ECCS Area Ventilation System is expected to be insignificant due to the low normal dose in the areas where the filters are located. The ECCS Area Ventilation System, located in the Auxiliary Building, is in a normally low dose area. No significant dose is received during surveillance activities in these areas of the Auxiliary Building. Therefore, the proposed amendment has no significant affect on either individual or cumulative occupational radiation exposure.

References

1. St. Lucie Unit No. 2 Updated Final Safety Analysis Report, through Amendment 17.
2. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.
3. Code of Federal Regulations, 10 CFR 50.67, "Accident Source Term," revised 12/03/02.
4. USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.
5. NEI 99-03, "Control Room Habitability Guidance," Nuclear Energy Institute, Revision 0 dated June 2001 and Revision 1 dated March 2003.
6. NAI-1101-044, "AST Licensing Technical Report for St. Lucie Unit 2," Revision 2, Numerical Applications, Inc., May 2007.
7. Code of Federal Regulations, 10 CFR 100.11, "Determination of exclusion area, low population zone, and population distance center."
8. Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.
9. Federal Guidance Report No. 12 (FGR 12), "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
10. Florida Power & Light Company, St. Lucie Unit No. 2 Technical Specifications (through Amendment 129).
11. Code of Federal Regulations, 10 CFR 50.92, "Issuance of Amendment."

12. Code of Federal Regulations, 10 CFR 51.22, "Criterion for Categorical Exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review."

13. L-2006-094, St. Lucie Unit 2 Proposed License Amendment Request, Steam Generator Tube Integrity, May 25, 2006.

14. NUREG-1432, Standard Technical Specifications Combustion Engineering Plants, Volume 1, Rev. 3.0

Table 1

St. Lucie Plant, Unit No. 2
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	500	1.16	2.56	4.46
MSLB – Outside of Containment (1.8% DNB)	435	0.33	0.91	4.69
MSLB – Outside of Containment (0.43% FCM)	435	0.37	0.98	4.77
MSLB – Inside of Containment (29% DNB)	435	0.54	1.09	4.96
MSLB – Inside of Containment (6.1% FCM)	435	0.79	1.50	4.92
SGTR Pre-accident Iodine Spike	500	0.25	0.24	2.57
Acceptance Criteria		**≤ 25[3]**	**≤ 25[3]**	**≤ 5[4]**
SGTR Concurrent Iodine Spike	500	0.06	0.06	0.66
Locked Rotor (13.7% DNB)	500	0.25	0.56	2.81
FWLB	500	0.02	0.02	0.82
Letdown Line Rupture	500	0.36	0.36	2.57
Acceptance Criteria		**≤ 2.5 [3]**	**≤ 2.5 [3]**	**≤ 5 [4]**
FHA - Containment	500	0.29	0.28	0.81
FHA – Fuel Handling Building	500	0.29	0.28	1.63
CEA Ejection – Containment Release (9.5% DNB, 0.5% FCM)	500	0.26	0.52	2.78
CEA Ejection – Secondary Release (9.5% DNB, 0.5% FCM)	500	0.30	0.65	2.87
Acceptance Criteria		**≤ 6.3[3]**	**≤ 6.3[3]**	**≤ 5[3]**

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

Technical Specification Change
Mark Ups

Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DEFINITIONS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/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)."~~

$\bar{E}$ - 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 RESPONSE 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. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.13 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.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Shield Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: At all times in MODES 1, 2, 3, and 4.
In addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6.

ACTION:

a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c.

b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable system to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(2) With both SBVS inoperable in MODE 1, 2, 3, or 4, immediately enter LCO 3.0.3.

c. (1) With one SBVS inoperable in any MODE, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

(2) With both SBVS inoperable in any MODE, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Shield Building Ventilation System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Performing a visual examination of SBVS in accordance with ~~ANSI N-510-1980.~~

ASME N510-1989

ASME N510-1989

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ~~ANSI N-510-1980.~~ The distribution shall be ± 20% of the average flow per unit.

~~3. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.~~

~~4. Verifying that the HEPA filter banks remove ≥ 99.825% of the DOP when they are tested in place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.~~

~~5. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N-510-1980.~~

c. ~~After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a 2 inch laboratory sample from the installed sample canisters demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).~~

d. At least once per 18 months by:

~~1. Verifying that the pressure drop across the demisters, electric heaters, HEPA filters, and charcoal adsorber banks is less than 8.5 inches Water Gauge (WG) while operating the system at a flow rate of 6000 cfm ± 10%.~~

2. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

3. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

~~4.~~ Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

~~5. Verifying that the main heaters dissipate 30 ± 3 kW and the auxiliary heaters dissipate 1.5 ± 0.25 kW when tested in accordance with ANSI N-510-1980.~~

By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

4

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

~~e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.825% of the DOP when they are tested in-place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.~~

PLANT SYSTEMS

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ECCS area ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ECCS area ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8 Each ECCS area ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by~~:~~

~~1.~~ ~~Verifying a system flow rate of 30,000 cfm ± 10% during system operation.~~ C

~~2.~~ Verifying that the system starts on a safety injection actuation test signal. V

b. By performing required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

ADMINISTRATIVE CONTROLS (Continued)

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 $< 0.60\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq$ ~~0.12~~ L_a for secondary containment bypass leakage paths.

0.096

b. Air lock testing acceptance criteria are:

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

2) For each door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions for T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

i. Inservice Testing Program

This program provides controls for Inservice testing of ASME Code Class 1, 2 and 3 components (pumps and valves). The program shall include the following:

a. Testing frequencies specified in Section XI of ASME Boiler and Pressure Vessel Code* and applicable addenda as follows:

ASME Boiler and Pressure Vessel Code* and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities.

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code* shall be construed to supersede the requirements of any technical specification.

* Where ASME Boiler and Pressure Vessel Code is referenced it also refers to the applicable portions of ASME/ANSI OM-Code, "Operation and Maintenance of Nuclear Power Plants," with applicable addenda, to the extent it is referenced in the Code.

ADMINISTRATIVE CONTROLS (continued)

j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a. a change in the TS incorporated in the license; or

   b. a change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.j.2.a or 6.8.4.j.2.b, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. Ventilation Filter Testing Program (VFTP)

~~A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.~~

REPLACE WITH INSERT 1

~~1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ 0.05% when tested in accordance with ANSI N510-1980 at the system flowrate specified below.~~

~~ESF Ventilation System~~	~~Flowrate~~
~~Control Room Emergency Air Cleanup~~	~~2000 ± 200 cfm~~

~~2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with ANSI N510-1980 at the system flowrate specified below.~~

~~ESF Ventilation System~~	~~Flowrate~~
~~Control Room Emergency Air Cleanup~~	~~2000 ± 200 cfm~~

~~3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.~~

~~ESF Ventilation System~~	~~Penetration~~	~~RH~~
~~Control Room Emergency Air Cleanup~~	~~≤ 0.175%~~	~~95%~~

k. Ventilation Filter Testing Program (VFTP) (continued)

~~4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.~~

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~
~~Control Room Emergency Air Cleanup~~	~~< 7.4" W.G.~~	~~2000 ± 200 cfm~~

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

l. Steam Generator (SG) Program

1. A SG Program shall be established and implemented for the replacement SGs to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following provisions:

   a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

   b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

      1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

      2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed ~~0.3~~ 0.5 gallons per minute total through all SGs and ~~216 gallons per day~~ 0.25 gallons per minute through any one SG.

      3. The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "Reactor Coolant System Operational Leakage."

INSERT 1

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 3.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Air Cleanup	≤ 0.05%	2000 ± 200 cfm
Shield Building Ventilation System	≤ 0.05%	6000 ± 600 cfm
ECCS Area Ventilation System	≤ 0.05%	30,000 ± 3000 cfm

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Air Cleanup	≤ 0.05%	2000 ± 200 cfm
Shield Building Ventilation System	≤ 0.05%	6000 ± 600 cfm
ECCS Area Ventilation System	≤ 0.05%	30,000 ± 3000 cfm

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Emergency Air Cleanup	≤ 0.175%	95%
Shield Building Ventilation System	≤ 2.5%	95%
ECCS Area Ventilation System	≤ 2.5%	95%

4. For the Control Room Emergency Air Cleanup System and the ECCS Area Ventilation System, demonstrate that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below. For the Shield Building Ventilation System, demonstrate that the pressure drop across the combined demisters, electric heaters, HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency Air Cleanup	< 7.4" W.G.	2000 ± 200 cfm
Shield Building Ventilation System	< 8.5" W.G.	6000 ± 600 cfm
ECCS Area Ventilation System	< 4.35" W.G.	30,000 ± 3000 cfm

5. At least once per 18 months, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	**Wattage**
Shield Building Ventilation System	
Main Heaters	30 $\pm$ 3 kW
Auxiliary Heaters	1.5 $\pm$ 0.25 k

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Changes

DEFINITIONS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/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 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 RESPONSE 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. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.13 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.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Shield Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: At all times in MODES 1, 2, 3, and 4.
In addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6.

ACTION:

a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c.

b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable system to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(2) With both SBVS inoperable in MODE 1, 2, 3, or 4, immediately enter LCO 3.0.3.

c. (1) With one SBVS inoperable in any MODE, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

(2) With both SBVS inoperable in any MODE, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Shield Building Ventilation System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Performing a visual examination of SBVS in accordance with ASME N510-1989.

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989. The distribution shall be ± 20% of the average flow per unit.

c. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

d. At least once per 18 months by:

1. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

2. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

3. Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

4. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

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PLANT SYSTEMS

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ECCS area ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ECCS area ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8 Each ECCS area ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room and verifying that the system operates for at least 15 minutes.

b. By performing required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

c. At least once per 18 months by verifying that the system starts on a safety injection actuation test signal.

ADMINISTRATIVE CONTROLS (Continued)

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 $< 0.60\ L_a$ 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 is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions for T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

l. Inservice TestingProgram

This program provides controls for inservice testing of ASME Code Class 1, 2 and 3 components (pumps and valves). The program shall include the following:

a. Testing frequencies specified in Section XI of ASME Boiler and Pressure Vessel Code* and applicable addenda as follows:

ASME Boiler and Pressure VesselCode* and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. Theprovisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities.

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code* shall be construed to supersede the requirements of any technical specification.

* Where ASME Boilerand Pressure Vessel Code is referenced it also refers to the applicable portions of ASME/ANSI OM-Code, "Operation and Maintenance of Nuclear Power Plants," with applicable addenda, to the extent it is referenced in the Code.

ADMINISTRATIVE CONTROLS (continued)

j Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a. a change in the TS incorporated in the license; or

   b. a change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.j.2.a or 6.8.4.j.2.b, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 3.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Air Cleanup	≤ 0.05%	2000 ± 200 cfm
Shield Building Ventilation System	≤ 0.05%	6000 ± 600 cfm
ECCS Area Ventilation System	≤ 0.05%	30,000 ± 3000 cfm

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Air Cleanup	≤ 0.05%	2000 ± 200 cfm
Shield Building Ventilation System	≤ 0.05%	6000 ± 600 cfm
ECCS Area Ventilation System	≤ 0.05%	30,000 ± 3000 cfm

ADMINISTRATIVE CONTROLS (continued)

k. Ventilation Filter Testing Program (VFTP) (continued)

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	**Penetration**	**RH**
Control Room Emergency Air Cleanup	≤ 0.175%	95%
Shield Building Ventilation System	≤ 2.5%	95%
ECCS Area Ventilation System	≤ 2.5%	95%

4. For the Control Room Emergency Air Cleanup System and the ECCS Area Ventilation System, demonstrate that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below. For the Shield Building Ventilation System, demonstrate that the pressure drop across the combined demisters, electric heaters, HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	**Delta P**	**Flowrate**
Control Room Emergency Air Cleanup	< 7.4" W.G.	2000 ± 200 cfm
Shield Building Ventilation System	< 8.5" W.G.	6000 ± 600 cfm
ECCS Area Ventilation System	< 4.35" W.G.	30,000 ± 3000 cfm

5. At least once per 18 months, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	**Wattage**
Shield Building Ventilation System	
Main Heaters	30 ± 3 kW
Auxiliary Heaters	1.5 ± 0.25 kW

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

l. Steam Generator (SG) Program

1. A SG Program shall be established and implemented for the replacement SGs to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following provisions:

   a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

ADMINISTRATIVE CONTROLS (continued)

l. Steam Generator (SG) Program (continued)

1. (continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gallons per minute total through all SGs and 0.25 gallons per minute through any one SG.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "Reactor Coolant System Operational Leakage."

TS Bases Changes
(Information Only)

SECTION NO.: 3/4.4	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 6 OF ADM-25.04 REACTOR COOLANT SYSTEM ST. LUCIE UNIT 2	PAGE: 6 of 15
REVISION NO.: 2		

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.4 PORV BLOCK VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs in conjunction with a reactor trip on a Pressurizer Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2, or 3.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Since it is impractical and undesirable to actually open the PORVs to demonstrate their reclosing, it becomes necessary to verify OPERABILITY of the PORV block valves to ensure capability to isolate a malfunctioning PORV. As the PORVs are pilot operated and require some system pressure to operate, it is impractical to test them with the block valve closed.

The PORVs are sized to provide low temperature overpressure protection (LTOP). Since both PORVs must be OPERABLE when used for LTOP, both block valves will be open during operation with the LTOP range. As the PORV capacity required to perform the LTOP function is excessive for operation in MODE 1, 2, or 3, it is necessary that the operation of more than one PORV be precluded during these MODES. Thus, one block valve must be shut during MODES 1, 2, and 3.

3/4.4.5 STEAM GENERATOR~~S~~ (SG) TUBE INTEGRITY

~~The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.~~

INSERT B3/4.4.5

~~Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.~~

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3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

(SG) TUBE INTEGRITY

3/4.4.5 STEAM ~~GENERATORS~~ (continued)

~~The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1.0 gpm from both steam generators. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.~~

~~Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.~~

INSERT B3/4.4.5

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3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. The LCO is consistent with NUREG-1432, Revision 1, and is satisfied when leakage detection monitors of diverse measurement means are OPERABLE in MODES 1, 2, 3, and 4. Monitoring the reactor cavity sump inlet flow rate, in combination with monitoring the containment particulate or gaseous radioactivity, provides an acceptable minimum to assure that unidentified leakage is detected in time to allow actions to place the plant in a safe condition when such leakage indicates possible pressure boundary degradation.

3/4.4.6.2 OPERATIONAL LEAKAGE

~~Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.~~

~~The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.~~

~~The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.~~

INSERT B3/4.4.6.2
(follows Insert for
B3/4.4.5)

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE (continued)

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

INSERT B3/4.4.6.2 (follows Insert for B3/4.4.5)

~~The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

~~PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.~~

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

0.25 gpm

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT B3/4.4.5

0.5

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 0.5 ~~0.3~~ gpm total and ~~0.16 gpd~~ through any one SG or is assumed to increase to ~~0.3~~ gpm total through all SGs and ~~0.16 gpd~~ through any one SG as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3), 10 CFR 50.67 (Ref. 7) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still have tube integrity. Tube repair (i.e., sleeving) is applicable only to the original SGs.

In the context of this Specification, a SG tube for the replacement SGs is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For the original SGs, when the alternate repair criteria in TS Section 6.8.4.l.2.c.4 are applied a SG tube is defined as the length of the tube, including the tube wall and any repairs made to it, between 10.3 inches below the bottom of the hot leg expansion transition or top of the tubesheet (whichever is lower) and the tube-to-tubesheet weld at the tube outlet. If a portion of a tube sleeve extends below 10.3 inches from the bottom of the hot leg expansion transition or the top of the tubesheet (whichever is lower) a SG tube is defined as the length of the tube between the bottom of the sleeve to the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.l, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT B3/4.4.5

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the *SG* tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is *defined* as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. *In that context, the term "significant" is defined as* "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, *other than a SGTR, is within the accident* analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed ~~0.3~~ 0.5 gpm total and ~~[illegible] gpd~~ 0.25 gpm through any one SG. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "*Reactor Coolant System operational leakage*," *and limits primary-to-secondary* leakage through any one SG to 150 gpd at room temperature. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT B3/4.4.6.2

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the sources of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

Applicable Safety Analyses 0.25 gpm 0.5

The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary leakage from all steam generators (SGs) is ~~0.3~~ gpm total through all SGs and ~~≤ 216 gpd~~ through any one SG or is assumed to increase to ~~0.3~~ gpm total through all SGs and ~~≤ 216 gpd~~ through any one SG as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd is based on room temperature conditions. When this value is adjusted for operating conditions, it is less than ~~or equal to~~ the leakage limit of ~~≤ 216 gpd~~ (measured at operating temperature) through any one SG assumed in the accident analysis. ~~St. Lucie Unit 2 procedures further administratively limit operational leakage with the intent that the accident induced leakage limits will not be exceeded.~~ e

0.25 gpm

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT B3/4.4.6.2

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

0.5 0.25 gpm

The FSAR (Ref. 5) analysis for SGTR assumes the contaminated secondary fluid is released mainly via the safety valves or atmospheric dump valves and only briefly steamed to the condenser. The 0.3 gpm total through all SGs and ~~0.16 gpd~~ through any one SG primary to secondary leakage safety analysis assumption is relatively inconsequential.

of 0.25

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes a value ~~greater than 0.15~~ gpm primary to secondary leakage through each generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in GDC 19, 10 CFR 100, 10 CFR 50.67 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. UNIDENTIFED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Primary-to-Secondary Leakage Through Any One Steam Generator

The limit of 150 gpd per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube

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3/4.4 **REACTOR COOLANT SYSTEM (continued)**

BASES (continued)

total primary-to-secondary leakage through all SGs and ~~216 gallons per day~~ through any one SG

0.25 gpm

3/4.4.8 **SPECIFIC ACTIVITY**

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed 10CFR50.67 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 0.5 ~~0.3~~ gpm and a loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation. R4

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take correction action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

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3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere and also reduces radioactive effluent releases to the environment during a fuel handling accident involving a recently irradiated fuel assembly in the spent fuel storage building. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

The fuel handling accident analysis assumes a minimum post reactor shutdown decay time of 72 hours. Therefore, recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. This represents the applicability bases for fuel handling accidents. Containment closure will have administrative controls in place to assure that a single normal or contingency method to promptly close the primary or secondary containment penetrations will be available. These prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw the release from the postulated fuel handling accident in the proper direction such that it can be treated and monitored.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

With respect to Surveillance 4.6.6.1.b, ~~Regulatory Guide 1.52, Revision 3, Section 6.3 states that testing is required "...following painting, fire or chemical release... that may have an adverse effect on the functional capability of the system." Additionally, Footnote 8 states the painting, fire, or chemical release is "not communicating with the HEPA filter or adsorber if the ESF atmosphere cleanup system is not in operation, the isolation dampers for the system are closed, and there is no pressure differential across the filter housing. This provides reasonable assurance that air is not passing through the filters and adsorbers." A program has been developed to control the use of paints and other volatiles in the areas served by the shield building ventilation system.~~

this SR verifies that the required Shield Building Ventilation System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP).

SECTION NO.: 3/4.7 REVISION NO.: 3	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 9 OF ADM-25.04 PLANT SYSTEMS ST. LUCIE UNIT 2	PAGE: 9 of 11

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the Control Room Emergency Air Cleanup System ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

With respect to Surveillance 4.7.7.c, Regulatory Guide 1.52, Revision 3, Section 6.3 states that testing is required "...following painting, fire or chemical release...that may have an adverse effect on the functional capability of the system." Additionally, Footnote 8 states the painting, fire, or chemical release is "not communicating" with the HEPA filter or adsorber if the ESF atmosphere cleanup system is not in operation, the isolation dampers for the system are closed, and there is no pressure differential across the filter housing. This provides reasonable assurance that air is not passing through the filters and adsorbers." A program has been developed to control the use of paints and other volatiles in the areas served by the control room emergency air cleanup system. R2

3/4.7.8 ECCS AREA VENTILATION SYSTEM

The OPERABILITY of the ECCS Area Ventilation System ensures that cooling air is provided for ECCS equipment.

INSERT → HERE

With respect to Surveillance 4.7.8.b, this SR verifies that the required ECCS Area Ventilation System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP).

NAI Report Release

Report Number: NAI-1101-044

Revision Number: 2

Title: AST Licensing Technical Report for St. Lucie Unit 2

Description:

This report documents the results of the analyses and evaluations performed by Numerical Applications, Inc. in support of the St. Lucie Unit 2 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.

Mark A Pope	5/24/07
Author (Mark Pope)	Date
Jim Harrell	5-24-07
Reviewer (Jim Harrell)	Date
Joe Sinodis	May 24, 2007
Reviewer (Joe Sinodis)	Date
T. George	May 24, 2007
NAI Management (Tom George)	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. 2, licensing basis for the radiological 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, with the exception of the Steam Generator Tube Rupture event which was revised to the Alternative Source Term (AST) methodology as part of License Amendment 138.

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, and supplemented by Regulatory Issue Summary 2006-04, is being used to calculate the offsite and control room radiological consequences for St. Lucie Unit No. 2 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
- Letdown Line Break
- Feedwater Line Break (FWLB)

Note that although RG 1.183 does not include the Letdown Line Break and Feedwater Line Break events, they were included in the AST analysis to provide consistency of methodology with the remaining limiting accident analyses.

Each accident and the specific input and assumptions are described in Section 2.0 of this report. The inputs and assumptions related to the steam generators are based on the replacement steam generators which are scheduled for installation during the fall 2007 refueling outage. These analyses provide for a bounding allowable control room unfiltered air inleakage of 435 cfm. The use of 435 cfm as a design basis value was established to be above the unfiltered inleakage value determined through testing and analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01.

1.3 Proposed Changes to the St. Lucie Unit No. 2 Licensing Basis

Florida Power and Light (FP&L) Company proposes to revise the St. Lucie Plant, Unit No. 2, licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences of the UFSAR Chapter 15 accidents listed in Section 1.2 above. As part of the full implementation of this AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10CFR50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11.
- New onsite (Control Room) and offsite atmospheric dispersion factors are developed.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.
- Increased values for control room unfiltered air inleakage are assumed.
- A steam generator tube leakage rate for accident analysis which exceeds the current Technical Specification operational leakage limit is utilized.
- Credit for the ECCS Area Ventilation System HEPA filters is being taken.
- 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. 2, 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," as the source of thyroid dose conversion factors.

- Surveillance Requirement 4.6.6.1 is revised to relocate the HEPA filter, charcoal adsorber, flow rate and heater surveillance test acceptance criteria for the Shield Building Ventilation System to the Ventilation Filter Testing Program in TS Section 6.8.4.k.

- Surveillance Requirement 4.7.8 is revised to relocate the flow rate surveillance test acceptance criterion for the ECCS Area Ventilation System to the Ventilation Filter Testing Program in TS Section 6.8.4.k.

- 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 12% to 9.6%.

- The HEPA filter and charcoal adsorber test acceptance criteria for the Shield Building Ventilation System relocated to the VFTP are revised as follows:
  - The HEPA filter efficiency test criterion is increased from 99.825% to 99.95%
  - The inplace charcoal adsorber efficiency test acceptance criterion is increased from 99% to 99.95%
  - The laboratory charcoal adsorber efficiency test acceptance criterion is increased from 90% to 97.5%

- The following ECCS Area Ventilation System HEPA filter and charcoal adsorber efficiency test criteria are added to the VFTP:
  - HEPA filter 99.95%
  - Inplace charcoal adsorber 99.95%
  - Laboratory charcoal adsorber 97.5%

- ECCS Area Ventilation System pressure drop test criteria are added to the VFTP.

- In the VFTP (TS 6.8.4.k), reference to Regulatory Guide 1.52, Revision 2 is replaced with reference to Regulatory Guide 1.52, Revision 3.

- In the Surveillance Requirements for the Shield Building Ventilation System (SR 4.6.6.1), as well as in the VFTP (TS 6.8.4.k), reference to ANSI N-510-1975 is replaced with reference to ASME N510-1989.

- The accident induced leakage performance criteria of the Steam Generator (SG) Program (TS Section 6.8.4.l) submitted for approval as part of the Steam Generator Tube Integrity License Amendment Request (Ref. 13) is changed to 0.5 gpm total through all SGs and 0.25 gpm through any one SG.

1.4 Compliance with Regulatory Guidelines

The revised St. Lucie Unit No. 2 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 Feedwater Line Break dose consequences acceptance criteria for the EAB and LPZ are based upon the guidance of Section 15.2.8 of the Standard Review Plan (SRP), which specifies that any activity release must be a small fraction of the 10CFR100 limits at the site boundary. A "small fraction" is interpreted elsewhere in the SRP to be less than ten percent, or 2.5 rem.

- Guidance for the evaluation of the Letdown Line Break event was obtained from Section 15.6.2 of the Standard Review Plan. The SRP for this event requires that an iodine spike be considered which increases the equilibrium fission product activity release rate from the fuel by a factor of 500. In addition, dose consequences acceptance criteria are established which limit exposures at the EAB and LPZ to within 10 percent of the 10CFR100 guidelines.

- 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	1.1a	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. 2, the events not specifically addressed in RG 1.183 are the Letdown Line Break and the Feedwater Line Break (FWLB).

1.6.3 Control Room Ventilation System Description

The Control Room Air Conditioning System (CRACS) and Control Room Emergency Cleanup System (CRECS) 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 Cleanup Systems are discussed in the St. Lucie Unit 2 UFSAR, Sections 6.4 and 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 1000 cfm). Following a design basis accident the control room is pressurized at the rate of up to 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 isolation valves close on loss of operating power. 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 Cleanup Systems is approximately 97,200 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, water-cooled 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 Cleanup 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. Redundant radiation monitors are located at both outside air intakes.

The Control Room Emergency Cleanup System (CRECS) consists of two redundant air cleaning trains. Each train consists of (in order of flow) a prefilter, a High Efficiency Particulate (HEPA) prefilter, a charcoal adsorber, a HEPA after filter and a centrifugal fan.

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 supplied through either of two outside air intakes located in the northern and southern walls of the Reactor Auxiliary Building at approximately elevation 78 feet. 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 automatically by the return dampers with their corresponding controller either in Auto or Manual control mode 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 CRECS 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 partial filtration of recirculated air, or

b) automatic isolation with immediate manual and/or automatic filtered pressurization and recirculation with partial 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 CRECS filtration units start automatically while the air conditioning units remain running. The isolation time including the instrument closure time is equal to a maximum, of 12 seconds, assuming offsite power is available. If offsite power is not available, the isolation time is 22 seconds, which includes the 10-second diesel generator start time (for dose analyses purposes, this value is conservatively increased to 30 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 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 CRECS 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 CRECS 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 CRECS fan is started.

The Control Room Emergency Cleanup System removes potentially radioactive particulates and iodine from the control room air during the post-LOCA operating mode. Each unit consists of a roughing filter, HEPA prefilter, charcoal adsorber, HEPA after-filter and fan. 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 redundant air cleaning units remove 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 2 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 1000 cfm. In this configuration, the dispersion factor for air being drawn into the control room is assumed to be the more limiting 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 up to 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 judgments 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.

For all events, delays in switching to the emergency/recirculation mode from the normal mode are conservatively considered 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 model imposes a 30-second delay to allow the CR ventilation system to physically switch into isolation mode. The time at which the operator will act to unisolate the control room and initiate the filtered air makeup is a proceduralized operator decision during the course of the event. For St. Lucie Unit 2, 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

Control room inleakage testing identified a potential unfiltered leakage pathway into the control room envelope via the switchgear room through louver 2L-11. Separate atmospheric dispersion factors were developed for this leakage pathway as described in Section 1.8. To ensure that the most liming configuration is considered, all events were analyzed using both the intake methodology described in Section 1.6.3 as well as with unfiltered inleakage through the switchgear rooms. Results of the limiting cases are presented in the discussion of the event analyses in Section 2.0, and the applicable release-receptor pair is shown in Table 1.8.1-3.

1.6.5 Direct Shine Dose

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

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

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

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. Per Table 6.4-2 of the UFSAR, all other sources of direct shine dose are considered negligible. The MicroShield 5 code is used to determine direct shine exposure to a dose point located in the control room. Each source required a different MicroShield case structure including different geometries, sources, and materials. The external cloud is assumed to have a length of 1000 meters in the MicroShield cases to approximate an infinite cloud. A series of cases is run with each structure to determine an exposure rate from the radiological source at given points in time. These sources were taken from RADTRAD-NAI runs that output the nuclide activity at a given point in time for the event. The RADTRAD-NAI output provides the time dependent results of the radioactivity retained in the control room filter components, as well as the activity inventory in the environment and the containment. A bounding CR filter inventory is established using a case from the sensitivity study with unfiltered inleakage that produced a control room dose slightly in excess of the 5 rem TEDE dose limit to control room operators. The direct shine dose calculated due to the filter loading for this conservative unfiltered inleakage case is used as a conservative assessment of the direct shine dose contribution for all accidents.

The RADTRAD-NAI sources were then input into the MicroShield case file 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 <u>Radiation Source Terms</u>

1.7.1 Fission Product Inventory

The source term data to be used in performing alternative source term (AST) analyses for St. Lucie Unit 2 are summarized in the following tables:

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

Note that the source terms provided in the referenced tables do not include any decay before the start of the events. Decay time assumptions are applied in the RADTRAD 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. 2 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 (2700 MW_{th} x 1.02 = 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 were obtained from 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.

Conservatisms used in the calculation of fission product inventories include the following:

- Use of ORIGEN 2.1 with revised data libraries for extended fuel burnup.
- Use of a core thermal power corresponding to the plant design power plus 2% calorimetric uncertainty.
- Use of bounding maximum assembly and bounding core average equilibrium cycle maximum burnup.
- Use of a bounding range of average assembly enrichments.
- Use of a bounding maximum assembly uranium loading.
- Neglect of decay of fission products during refueling outages

1.7.2 Primary Coolant Source Term

The primary coolant source term for St. Lucie Unit 2 is derived from Table 11.1-2 of the UFSAR. Per the assumptions listed in Table 11.1-1 of the UFSAR, the activities given in Table 11.1-2 are based on

1% failed fuel. Table 11.1-2 of 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 11.1-2 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 thyroid values from FGR 11 for iodine isotopes. Dose equivalent I-131 calculated using the thyroid dose conversion factors results in higher primary coolant concentrations than the DE I-131 determined from effective DCFs. The non-iodine species are adjusted to achieve the Technical Specification limit of 100/E-bar microcuries per gram of gross activity for non-iodine activities.

The dose conversion factors for inhalation and submersion are from Federal Guidance Reports Nos. 11 and 12 respectively. The final adjusted primary coolant source term is presented in Table 1.7.2-1, "Primary Coolant Source Term."

1.7.3 Secondary Side Coolant Source Term

Secondary coolant system activity is limited to a value of ≤ 0.10 µCi/gm dose equivalent I-131 in accordance with TS 3.7.1.4. Noble gases entering the secondary coolant system are assumed to be immediately released; thus the noble gas activity concentration in the secondary coolant system is assumed to be 0.0 µCi/gm. Thus, the secondary side iodine activity is 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 Reg. Guide 1.183, the inventory of fission products in the St. Lucie Unit 2 reactor core available for release to the containment are based on the maximum full power operation of the core 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 2 assumes the failure of one assembly; therefore, the fuel handling accident source term is based on a single "bounding" fuel assembly.

Per Section 3.1 of Reg. Guide 1.183, the source term methodology for the Fuel Handling Accident is similar to that used for developing the LOCA containment leakage source term, except that for DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, a radial peaking factor of 1.7 (total integrated peaking factor per St. Lucie Unit 2 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 at an average assembly power of 12.691 MW_{th}. Thus, based on the methodology specified in Reg. Guide 1.183, the fuel handling accident source term is derived by applying a factor of 1.7/217 to the LOCA containment leakage source term, where 1.7 is the radial peaking factor. 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 Handlir.g Accident Source Term."

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

1.8.1 Onsite *X/Q* Determination

New *X/Q* factors for onsite release-receptor combinations are developed using the ARCON96 computer code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664). Additionally, NRC Regulatory Guide 1.94, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003, has been implemented. Reg. Guide 1.94 contains new guidance that supersedes 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.

Figure 1.8.1-1 provides a sketch of the general layout of St. Lucie Unit 2 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 Assessments 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 and for switchgear room louver 2L-11 that apply to the various accident scenarios for onsite control room dose consequence analyses. For the intakes, values are presented for the unfavorable intake prior to control room isolation, the midpoint between the intakes during isolation, as well as values for the favorable intake following manual restoration of filtered control room makeup flow. These values include credit for dilution where allowed by Reg. Guide 1.194. 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 Basis Accidental 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 selected for use in the evaluations. Note that the 0-2 hour EAB atmospheric dispersion factor is applied to all time periods in the analyses.

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 2 containment is given as 225.50 ft. The highest possible release point is not 2.5 times higher than the adjacent containment building; therefore, all releases are considered ground level releases. As such, the release height is set equal to 10.0 meters as required by Table 3.1 of NUREG/CR-2858. The building area used for the building wake term is the same as for 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 ft.

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) are used in the development of the new *X/Q* factors used in the analysis. The St. Lucie Plant, Unit No. 2, Meteorological Monitoring Program, complies with RG 1.23; "Onsite Meteorological Programs," February 1972. The Meteorological Monitoring Program is described in Section 2.3.3 of the St. Lucie Plant Unit No. 2 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.

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 2 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 Reg. Guide 1.194 and NUREG/CR-6331 on the valid meteorological data recovery rate required for use in determining onsite *X/Q* values. However, Regulatory Position C.5 of RG 1.23 requires a 90% data recovery threshold for measuring and capturing meteorological data. Clearly, the 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 ARCON96 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 15.6.5 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 - Per UFSAR Fig. 6.5-8, the long term recirculation sump pH remains greater than 7.0. 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 iodine. 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 both the sprayed and unsprayed regions of containment.

   A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region of containment as well as in the sprayed region after spray is terminated at 8 hours. Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," December 1983, documents results from Containment Systems Experiment testing. These tests show that settling of aerosols due to gravity is the dominant natural mechanism for fission product retention. This report finds that significant removal by sedimentation would be expected even at very low particulate concentrations. Figure 4-2 of IDCOR Program Technical Report 11.3 shows a ten-fold reduction in the airborne cesium concentration over a 7-hour period at relatively low concentrations. This represents an aerosol removal rate of 0.33 hr^{-1}. A more conservative value of 0.1 hr^{-1} is used in the analyses based upon NRC approval of this value in the safety evaluations for the Harris Nuclear Plant License Amendment No. 107 in October 2001 (ADAMS Accession No. ML012830516) and the

Kewaunee Nuclear Power Plant License Amendment No. 166 in March 2003 (ADAMS Accession No. ML030210062).

No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 - Containment spray provides coverage to 85% of the containment. Therefore, the St. Lucie Unit 2 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.

   Reg. Guide 1.183 and the SRP state that the elemental iodine spray removal coefficient should be set to zero when a decontamination factor (DF) of 200 is reached for elemental iodine. The particulate spray removal coefficient should be reduced by a factor of 10 when a DF of 50 is reached for the aerosol.

   As discussed in the SRP, the iodine decontamination factor (DF) is a function of the effective iodine partition coefficient between the sump and containment atmosphere. Thus, the loss of iodine due to other mechanisms (containment leakage, surface deposition, etc.), would not be included in the determination of the time required to reach a DF of 200. In addition, since the iodine in the containment atmosphere and sump are decaying at the same rate, decay should not be included in determining the time to reach a DF of 200. Additional RADTRAD-NAI cases were performed for determining the time to reach a decontamination factor of 200.

   The first RADTRAD-NAI case was used to determine the peak containment atmosphere elemental iodine concentration and amount of aerosol in the containment atmosphere. This case included:

- No containment spray
- No surface deposition
- No decay
- No containment leakage

The second RADTRAD-NAI case determined the time required to reach a DF of 200 based on the peak elemental iodine concentration from the first RADTRAD-NAI case. The second RADTRAD-NAI case included:

- Containment sprays actuated at 0.016667 hours (60 seconds)
- No surface deposition
- No decay
- No containment leakage

The second RADTRAD-NAI case showed that a DF of 200 for elemental iodine was reached at a time greater than 3.06 hours.

A third RADTRAD-NAI case determined the time required to reach a DF of 50 for aerosol based on the peak aerosol mass from the first RADTRAD-NAI case. The third RADTRAD-NAI case included:

- Containment sprays actuated at 0.016667 hours (60 seconds)
- Aerosol surface deposition credited
- No decay
- No containment leakage

The third RADTRAD-NAI case showed that a DF of 50 was reached at a time greater than 2.65 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 - This position relates to suppression pool scrubbing in BWRs, which is not applicable to St. Lucie Unit No. 2.

8. Regulatory Position 3.6 - This position relates to activity retention in ice condensers, which is not applicable to St. Lucie Unit No. 2.

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. 100% of the radionuclide inventory of the RCS is released instantaneously at the beginning of the event. The containment purge flow is 2500 cfm through the eight-inch line and is assumed to be isolated after 30 seconds. No filters are credited.

11. Regulatory Position 4.1 - Leakage from containment collected by the secondary containment is processed by ESF filters prior to release from the plant stack.

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 310 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 as a ground level release 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. The filters in the SBVS ventilation system are credited at 99% efficiency for particulates and 95% for both elemental and organic iodine.

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 greater than two times the value identified in UFSAR Section 6.5.1.3 for pump seals and valve stems in the ECCS area. The leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continue for the 30-day duration. Backleakage to the RWT is also considered separately as two times 1 gpm, which is bounding value based upon RCS leakage monitoring documented in the Control Room database. RWT leakage is assumed to begin at the start of recirculation and continue for the remainder of 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 - A flashing fraction of 3.4% was calculated based upon the sump temperature at the time of recirculation. 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 - For ECCS leakage into the auxiliary building, the form of the released iodine is 97% elemental and 3% organic. An ECCS area ventilation system filter efficiency of 95% is assumed for both elemental and organic iodine. The ECCS area ventilation system meets the requirements of RG 1.52 and Generic Letter 99-02. There is no credit for hold-up or dilution in the Reactor Auxiliary Building.

    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 dilution of the activity in the RWT.

23. Regulatory Position 6 - This position relates to MSSV leakage in BWRs, which is not applicable to St. Lucie Unit No. 2.

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

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 (high containment pressure) is reached. The following measures will limit the consequences of the accident in two ways:

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

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

Release Inputs

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

From TS Surveillance Requirement 3.6.1.1, the initial leakage rate from containment is 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 1.28 gph based upon two times the current licensing basis value of 0.64 gph. The leakage is conservatively assumed to start at 20 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. The form of the released iodine is 97% elemental and 3% organic. Dilution and holdup of the ECCS leakage in the Reactor Auxiliary Building are not credited.

The ECCS backleakage to the RWT is initially assumed to be 2 gpm based upon doubling the current bounding value of 1 gpm. This leakage is assumed to start at 20 minutes into the event when recirculation starts and continue throughout the 30-day period. Based on sump pH history, the iodine in the sump solution is assumed to all be nonvolatile. However, when introduced into the acidic solution of the RWT inventory, there is a potential for the particulate iodine to convert into the elemental form. The fraction of the total iodine in the RWT which becomes elemental is both a function of the RWT pH and the total iodine concentration. The amount of elemental iodine in the RWT fluid which then enters the RWT air space is a function of the temperature-dependent iodine partition coefficient.

The time-dependent concentration of the total iodine in the RWT (including stable iodine) was determined from the tank liquid volume and leak rate. This iodine concentration ranged from a minimum value of 0 at the beginning of the event to a maximum value of 3.7E-05gm-atom/liter at 30 days (see Table 2.1-4). Based upon the backleakage of sump water, the RWT pH slowly increases from an initial value of 4.9 to a maximum pH of 5.3 at 30 days (see Table 2.1-3). Using the time-dependent RWT pH and the total iodine concentration in the RWT liquid space, the amount of iodine converted to the elemental form was determined using guidance provided in NUREG-5950. This RWT elemental iodine fraction ranged from 0 at the beginning of the event to a maximum of 0.047 (see Table 2.1-6).

The elemental iodine in the liquid region of the RWT is assumed to become volatile and to partition between the liquid and vapor space in the RWT based upon the partition coefficient for elemental iodine as presented in NUREG-5950. A GOTHIC model was used to determine the RWT temperature as a function of time (see Table 2.1-5) which was then used to calculate the partition coefficient shown in Table 2.1-7. The RWT is a vented tank; therefore, there will be no pressure transient in the air region that would affect the partition coefficient. 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. The elemental iodine flow rate from the RWT is equal to the air flow rate times the elemental iodine concentration in the RWT vapor space.

For the organic iodine flow, the same approach was used with an organic iodine fraction of 0.0015 from Reg. Guide 1.183 in combination with a partition coefficient of 1.0. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWT. Therefore, the total iodine flow is the sum of the elemental and organic iodine flow rates.

The time dependent iodine release rate presented in Table 2.1-8 in then applied to the entire iodine inventory (particulate, elemental and organic) in the containment sump. The iodine released via the RWT air vent to the environment was effectively set to 100% elemental (the control room filters have the same efficiency for all forms of iodine).

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 at 2500 cfm until isolation occurs.

The release point for each of the above sources is presented in Table 1.8.1-3.

Transport Inputs

During the LOCA event, the initial containment purge is released through the plant stack with no filtration. 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 310 seconds. Activity subsequently collected by the SBVS is assumed to be a filtered release from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The activity that bypasses the SBVS is released unfiltered to the environment via a ground level release from containment. ECCS leakage into the Auxiliary Building is modeled as a release via the Auxiliary Building. For this release path, the ECCS area ventilation system is credited with a particulate removal efficiency of 99% and elemental and organic iodine efficiencies of 95%. 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 1000 cfm of unfiltered fresh air and an assumed value of 500 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a CIAS as a result of a high containment pressure signal. A 30-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, 500 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 up to 450 cfm of filtered makeup flow, 500 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, elemental iodine, and organic iodine.

LOCA Removal Inputs

Reduction of the airborne radioactivity in the containment by natural deposition is credited. The natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.89 hr^{-1}. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region and in the sprayed region after spray flow is secured at 8 hours. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 85% of the containment. Therefore, the St. Lucie Unit 2 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 elemental spray coefficient is limited to 20 hr^{-1} per SRP 6.5.2. This coefficient is reduced to 0 when an elemental decontamination factor (DF) of 200 is reached. Based upon the elemental iodine removal rate of 20 hr^{-1}, the DF of 200 is conservatively computed to occur at 3.06 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 6.40 hr^{-1}, the DF of 50 is conservatively computed to occur at 2.65 hours.

2.1.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. 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 30 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.

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.

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 15.7.4.1.2 of the UFSAR, which specifies that all of the fuel rods in a single fuel assembly are damaged.

This analysis considers both a dropped fuel assembly inside the containment with the maintenance hatch open, and an assembly drop inside the FHB. Section 9.4.2.2.2 of the UFSAR identifies that in the event of a high radiation signal in the FHB, the pool area is exhausted to the plant vent via the SBVS filters. However, a more conservative approach is taken for the drop inside in the FHB, and the release is assumed to occur directly from the FHB without filtration. 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.

A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and FHB release locations. This water level ensures an elemental iodine decontamination factor of 285 per the guidance provided in NRC Regulatory Issue Summary 2006-04.

2.2.2 Compliance with RG 1.183 Regulatory Positions

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

1. Regulatory Position 1.1 - The amount of fuel damage is assumed to be all of the fuel rods in a single fuel assembly per UFSAR Section 15.7.4.1.2.

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 285 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. Guidance for the use of 285 for the elemental iodine decontamination factor is provided in NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms."

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. 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. Iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. All activity released from the pool is assumed to leak to the environment over a two-hour period. No credit for dilution in the containment or FHB is taken.

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. The analysis includes a decay time of 72 hours before the beginning of fuel movement. Since the FHA source term presented in Table 1.7.5-1 does not include this decay time, it is accounted for in the RADTRAD-NAI model.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air and an assumed value of 500 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 500 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 up to 450 cfm of filtered makeup flow, 500 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, elemental iodine, and organic iodine.

2.2.4 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. For the FHA event, the release from the FHB is the closest point to the control room. The containment release corresponds to the containment maintenance hatch.

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 30 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 for the different modes of control room operation during this 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.

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. The affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to either the environment or the containment. Plant cool down is achieved via the remaining unaffected SG. This event is described in the UFSAR, Section 15.1.

2.3.2 Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis followed 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.

3. Regulatory Position 2 - Fuel damage is assumed for this event. It was determined that the activity released from the damaged fuel will exceed that released by the two iodine spike cases; therefore, the two iodine spike cases were not analyzed.

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

5. Regulatory Position 4 - 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.

6. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by proposed TS 6.8.4.l (0.5 gpm total, 0.25 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.

7. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.

8. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F.

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

10. 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, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

11. Regulatory Position 5.5.2 - The 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.

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

13. 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.

14. Regulatory Position 5.6 - Steam generator tube bundle uncovery in the intact SG is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 2. During this period, the fraction of primary-to-secondary leakage which 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. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.3.3 Other Assumptions

1. The initial RCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the intact SG are provided in Table 2.3-2. These values are based upon a cooldown rate of 100 °F/hr until the RCS temperature reaches 300 °F. This temperature is maintained until 8 hours when Shutdown Cooling is assumed to become available. The cooldown is then continued at a rate of 38 °F/hr until the RCS temperature is reduced to 212 °F at 10.32 hours. No credit is taken for energy removal by the Shutdown cooling system.

3. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 2.3-3.

4. 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).

5. For the purposes of determining the iodine concentration of the SG secondary, the mass in the unaffected SG is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore water level above the top of the tubes in the unaffected steam generator within a conservative time of one hour following a reactor trip.

6. All secondary releases are postulated to occur from the ADV with the most limiting atmospheric dispersion factors. Releases from containment through the SBVS are assumed to be released from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The activity that bypasses the SBVS is released unfiltered to the environment via a ground level release from containment.

7. The initial leakage rate from containment is 0.5% of the containment air per day. This leak rate is reduced by 50% after 24 hours 0.25% /day. 9.6% of the containment leakage is assumed to bypass the SBVS filters.

8. For the MSLB inside of containment, natural deposition of the radionuclides is credited consistent with the LOCA methodology presented in Section 2.1.3. 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 is based upon a double-ended break of one main steam line outside of containment, and the second case is based upon a double-ended break of one main steam line inside of containment. The primary difference between these two models is the transport of the primary-to-secondary leakage through the affected steam generator. Upon a MSLB, the affected SG rapidly depressurizes. 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 for both cases assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. The source term for this activity is presented in Table 1.7.4-1 with adjustments for the fraction of damaged fuel, the non-LOCA fission product gap fractions from Table 3 of RG 1.183, and a radial peaking factor of 1.7. All noble gases associated with this leakage are assumed to be released directly to the environment. For the break outside containment, primary-to-secondary leakage into the affected steam generator is also assumed to directly enter the atmosphere. For the break inside containment, the affected steam generator leakage is released into containment. All primary-to-secondary leakage is assumed to continue until the faulted steam generator is completely isolated at 12 hours.

Primary-to-secondary tube leakage is also postulated to occur in the unaffected SG for both cases. This activity is diluted by the contents of the steam generator and released via steaming from the ADVs until the RCS is cooled to 212°F. In addition, the analysis of both cases assumes that the initial iodine activity of both SGs is released directly to the environment. The entire contents of the faulted steam generator is released immediately, while the intact steam generator release occurs during the RCS cooldown. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 µCi/gm DE I-131 permitted by Tech. Specs. 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.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air and an assumed value of 435 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 435 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 up to 450 cfm of filtered makeup flow, 435 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, elemental iodine, and organic iodine.

2.3.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. For the MSLB event, all secondary releases are from the closest ADV. X/Qs for containment releases via the SBVS are from the plant stack, and containment releases which bypass the SBVS correspond to the nearest feedwater line which penetrates containment.

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 30 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 for the different modes of control room operation during this 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.

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 2.3-4, 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. 2 SGTR event. This event is described in Section 15.6.3 of the UFSAR.

The dose consequence analysis for this event with AST was previously approved in License Amendment No. 138. This event is re-analyzed with the following changes to the analysis assumptions:

- o Primary-to-secondary steam generator leakage is increased from 0.15 gpm/SG to 0.25 gpm/SG.
- o Steam releases to the atmosphere are increased to bound the replacement steam generators.
- o Intact steam generator tube uncovery is postulated for up to 45 minutes.
- o Releases through the intact steam generator ADV are assumed to continue beyond shutdown cooling conditions until the RCS temperature reaches 212 °F.

2.4.2 Compliance with RG 1.183 Regulatory Positions

The revised SGTR dose consequence analysis follows 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. No fuel damage is postulated to occur for the St. Lucie Unit No. 2 SGTR event.

2. Regulatory Position 2 - No fuel damage is postulated to occur for the St. Lucie Unit No. 2 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 6.8.4.l (0.5 gpm total, 0.25 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 based upon RCS conditions, consistent with the plant design basis.

9. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F. 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 - 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. For the unaffected steam generator used for plant cooldown, flashing is considered immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes 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 in the intact SG is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 2. During this period, the fraction of primary-to-secondary leakage which 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. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.4.3 Other Assumptions

1. This evaluation assumes that the RCS mass remains constant throughout the event

2. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore water level above the top of the tubes in the unaffected steam generator within a conservative time of one hour following a reactor trip.

3. Data used to calculate the iodine equilibrium appearance rates 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 concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum, 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 at 30 minutes.

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 steaming from the ADVs until the RCS is cooled to 212°F.

Per the St. Lucie Unit 2 UFSAR, Section 15.6.3.5.1, 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 RCS is cooled below 212°F. 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. Parameters used in the determination of the iodine equilibrium release rate are provided in Table 2.4-4. The iodine activities and the appearance rates for the accident-induced (concurrent) iodine spike case are presented in Table 2.4-5. All other release assumptions for this case are identical to those for the pre-accident spike case.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air and an assumed value of 500 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. For this event, it is conservatively assumed that the CR isolation signal is delayed until the release from the ADVs is initiated at 379.2 seconds. An additional 30-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, 500 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 up to 450 cfm of filtered makeup flow, 500 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, elemental iodine, and organic iodine.

2.4.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. Prior to reactor trip, the release is assumed to originate from the condenser. Following the trip, the releases from both the intact and faulted SGs are assumed to occur from the closest ADV.

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 30 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 for the different modes of control room operation during this 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.

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 2 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 15.3.3 of the UFSAR.

2.5.2 Compliance with RG 1.183 Regulatory Positions

The Locked Rotor dose consequence analysis followed 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 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 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 SGs as specified by proposed TS 6.8.4.l (0.5 gpm total, 0.25 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.

6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.

7. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F.

8. Regulatory Position 5.4 - The analysis assumes a coincident loss of offsite power in the evaluation of fission products released from the secondary system.

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 - 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 are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes 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 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 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 in the SGs is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 2. During this period, the fraction of primary-to-secondary leakage which 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. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

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-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

3. The steam mass release rates for the SGs are provided in Table 2.5-2. These values are based upon a cooldown rate of 100 °F/hr until the RCS temperature reaches 300 °F. This temperature is maintained until 8 hours when Shutdown Cooling is assumed to become available. The cooldown is then continued at a rate of 38 °F/hr until the RCS temperature is reduced to 212 °F at 10.32 hours. No credit is taken for energy removal by the Shutdown cooling system.

4. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 2.5-3.

5. This evaluation assumes that the RCS mass remains constant throughout the event.

6. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore secondary water level above the top of the tubes within a conservative time of one hour following a reactor trip.

7. 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 to experience DNB. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant. The source term is based upon release fractions from Appendix G of RG 1.183 with a radial peaking factor of 1.7. 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 until the RCS is cooled to 212°F. These release assumptions are consistent with the requirements of RG 1.183.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air and an assumed value of 500 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 500 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 up to 450 cfm of filtered makeup flow, 500 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, elemental iodine, and organic iodine.

2.5.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. For the Locked Rotor event, all releases are from the closest ADV.

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 30 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 for the different modes of control room operation during this 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.

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 2.5-4, 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 performed using steam release from the SG 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 15.4.8.

2.6.2 Compliance with RG 1.183 Regulatory Positions

The CEA Ejection dose consequence analysis followed 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 6.8.4.l (0.5 gpm total, 0.25 to any one SG).

9. Regulatory Position 7.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.

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 - Regulatory Position 7.4 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 - For the secondary release case, both steam generators are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes 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 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 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 in the SGs is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 2. During this period, the fraction of primary-to-secondary leakage which 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. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.6.3 Other Assumptions

1. The initial RCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the SGs are provided in Table 2.6-2. These values are based upon a cooldown rate of 100 °F/hr until the RCS temperature reaches 300 °F. This temperature is maintained until 8 hours when Shutdown Cooling is assumed to become available. The cooldown is then continued at a rate of 38 °F/hr until the RCS temperature is reduced to 212 °F at 10.32 hours. No credit is taken for energy removal by the Shutdown cooling system.

3. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 2.6-3.

4. This evaluation assumes that the RCS mass remains constant throughout the event.

5. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore secondary water level above the top of the tubes within a conservative time of one hour following a reactor trip.

6. Following the CEA Ejection event, 9.5% of the fuel is assumed to fail as a result of DNB and 0.05% of the fuel is assumed to experience fuel centerline melt.

7. All secondary releases are postulated to occur from the ADV with the most limiting atmospheric dispersion factors. Releases from containment through the SBVS are assumed to be released from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The activity that bypasses the SBVS is released unfiltered to the environment via a ground level release from containment.

8. The initial leakage rate from containment is 0.5% of the containment air per day. This leak rate is reduced by 50% after 24 hours 0.25% /day. 9.6% of the containment leakage is assumed to bypass the SBVS filters.

9. For the release inside of containment, natural deposition of the radionuclides is credited consistent with the LOCA methodology presented in Section 2.1.3. Containment sprays are not credited.

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 ADVs until the RCS is cooled to 212°F. All noble gases associated with this leakage are assumed to be released directly to the environment. The primary-to-secondary leakage is assumed to continue until the faulted steam generator is completely isolated at 12 hours. In addition, the analysis assumes that the initial iodine activity of both SGs is immediately released to the environment. The secondary coolant iodine concentration is assumed to

be the maximum value of 0.1 µCi/gm DE I-131 permitted by Tech. Specs. 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 1000 cfm of unfiltered fresh air and an assumed value of 500 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 500 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 up to 450 cfm of filtered makeup flow, 500 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, elemental iodine, and organic iodine.

2.6.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. For the CEA Ejection event, all secondary releases are from the closest ADV. X/Qs for containment releases via the SBVS are from the plant stack, and containment releases which bypass the SBVS correspond to the nearest feedwater line which penetrates containment.

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 30 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 for the different modes of control room operation during this 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.

The radiological consequences of the CEA Ejection are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.6-4, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.7 Letdown Line Rupture

2.7.1 Background

This event is a rupture of a primary coolant letdown line outside of containment. In accordance with the assumptions of UFSAR Section 15.6.2, the dose assessment for this event is comprised of a double ended rupture of the letdown line in the auxiliary building outside of containment with a direct release to the environment via the plant stack.

2.7.2 Compliance with RG 1.183 Regulatory Positions

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

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

1. Regulatory Position 5.6 of Appendix A - For ECCS leakage into the auxiliary building, the form of the released iodine is 97% elemental and 3% organic.

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

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

4. Regulatory Position 5.1 of Appendix E - The primary-to-secondary leak rate is apportioned between the SGs as specified by proposed TS 6.8.4.l (0.5 gpm total, 0.25 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.

5. Regulatory Position 5.2 of Appendix E - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.

6. Regulatory Position 5.3 of Appendix E - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F.

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

8. Regulatory Position 5.5.1 of Appendix E - Both steam generators are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

9. Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes 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.

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

11. Regulatory Position 5.5.4 of Appendix E - 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 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%.

12. Regulatory Position 5.6 of Appendix E - Steam generator tube bundle uncovery in the SGs is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 2. During this period, the fraction of primary-to-secondary leakage which 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. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.7.3 Other Assumptions

1. The initial RCS activity is assumed to be at the TS limit of 1.0 µCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 µCi/gm Dose Equivalent I-131.

2. The steam mass release values for the intact SG are provided in Table 2.7-2. These values are based upon a cooldown rate of 100 °F/hr until the RCS temperature reaches 300 °F. This temperature is maintained until 8 hours when Shutdown Cooling is assumed to become available. The cooldown is then continued at a rate of 38 °F/hr until the RCS temperature is reduced to 212 °F at 10.32 hours. No credit is taken for energy removal by the Shutdown cooling system.

3. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 2.7-5.

4. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore secondary water level above the top of the tubes within a conservative time of one hour following a reactor trip.

2.7.4 Methodology

This event is a rupture of a primary coolant letdown line outside of containment. In accordance with the assumptions of UFSAR Section 15.6.2, the dose assessment for this event is comprised of a double ended rupture of the letdown line in the auxiliary building outside of containment with a direct release to the environment via the plant stack. Additional release also occurs as a result of secondary side steam relief following the turbine trip and subsequent plant cooldown. Since RG 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis.

In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike. 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 500 times the iodine equilibrium release rate for a period of 8 hours. Parameters used in the determination of the iodine equilibrium release rate are provided in Table 2.7-3. The iodine activities and the appearance rates are shown in Table 2.7-4.

The Letdown Line Rupture flow rate is modeled as 85,788 lb_m over 1920 seconds with a flashing fraction of 25.9% as computed using the RG 1.183 guidance from position 5.4 of Appendix A for ECCS leakage. All of the activity in the flashed fluid is assumed to be released directly to the environment. Additional activity, based on the proposed primary-to-secondary leakage limits, is released via steaming from the ADVs until the RCS is cooled to 212°F.

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 1000 cfm of unfiltered fresh air and an assumed value of 500 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 500 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 up to 450 cfm of filtered makeup flow, 500 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, elemental iodine, and organic iodine.

2.7.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. For the Letdown Line Break event, the secondary releases are from the closest ADV and the activity released from the leak RCS fluid is released from the plant stack.

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 30 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 for the different modes of control room operation during this 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.

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

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

The radiological consequences of the Letdown Line Rupture event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 2.7-6, "Letdown Line Rupture Dose Consequences," the radiological consequences of the Letdown Line Rupture event are all within the appropriate acceptance criteria.

2.8 Feedwater Line Break (FWLB)

2.8.1 Background

This event involves a rupture of a main feedwater system pipe during plant operation. The rupture rapidly reduces the steam generator secondary inventory causing a partial loss of secondary heat sink, thereby allowing the heat-up of the Reactor Coolant System (RCS). A loss of offsite power occurs at the time of the trip. Plant cooldown is achieved via the remaining unaffected steam generator. No fuel failures are postulated to occur as a result of this event. This event is currently presented in Section 15.2.8 of the St. Lucie Unit 2 UFSAR.

2.8.2 Compliance with RG 1.183 Regulatory Positions

Since Regulatory Guide 1.183 does not provide any direct guidance regarding analysis of a Feedwater Line Break, Standard Review Plan (SRP) Section 15.8.2 is used as the primary source of guidance for this analysis. In addition, The RG 1.183 guidance provided for other events is applied to this event as applicable and appropriate. Therefore, the FWLB event dose consequence analysis is performed following the guidance provided in RG 1.183, as discussed below:

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

2. Regulatory Position 4 of Appendix E - Iodine releases from the faulted SG and the unaffected SG to the environment are assumed to be 97% elemental and 3% organic.

3. Regulatory Position 5.1 of Appendix E - The primary-to-secondary leak rate is apportioned between the SGs as specified by proposed TS 6.8.4.1 (0.5 gpm total, 0.25 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.

4. Regulatory Position 5.2 of Appendix E - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.

5. Regulatory Position 5.3 of Appendix E - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F.

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

7. Regulatory Position 5.5.1 of Appendix E - In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected steam generator used for plant cooldown, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

8. Regulatory Position 5.5.2 of Appendix E - The postulated leakage that immediately flashes 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.

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

10. Regulatory Position 5.5.4 of Appendix E - 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 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%.

11. Regulatory Position 5.6 of Appendix E - Steam generator tube bundle uncovery in the unaffected SG is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 2. During this period, the fraction of primary-to-secondary leakage which 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. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.8.3 Other Assumptions

1. The feedwater line break is assumed to be located outside of containment resulting in a blowdown of the affected generator to atmosphere from the most limiting release location (i.e., highest atmospheric dispersion factor along the feedwater line outside of containment).

2. The initial RCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

3. The steam mass release values for the intact SG are provided in Table 2.8-2. These values are based upon a cooldown rate of 100 °F/hr until the RCS temperature reaches 300 °F. This temperature is maintained until 8 hours when Shutdown Cooling is assumed to become available. The cooldown is then continued at a rate of 38 °F/hr until the RCS temperature is reduced to 212 °F at 10.32 hours. No credit is taken for energy removal by the Shutdown cooling system.

4. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 2.8-3.

5. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore secondary water level above the top of the tubes within a conservative time of one hour following a reactor trip.

2.8.4 Methodology

Input assumptions used in the dose consequence analysis of the FWLB are provided in Table 2.8-1. The analysis assumes that the entire fluid inventory from the faulted SG is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DE I-131 permitted by Tech. Specs. Additional activity due to primary-to-secondary leakage into

the faulted SG is also released directly to the environment. The source term for this activity is presented in Table 1.7.3-1. This leakage continues until the affected steam generator is completely isolated at 12 hours.

Primary-to-secondary tube leakage is also postulated to occur in the unaffected SG. This activity is diluted by the contents of the steam generator and released via steaming from the ADVs, along with the initial iodine activity of unaffected SGs. All releases from the unaffected steam generator continue until the RCS is cooled to 212 °F.

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

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air and an assumed value of 500 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation signal from the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 500 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 as per procedures. During this operational mode, the air flow distribution consists of up to 450 cfm of filtered makeup flow, 500 cfm of unfiltered inleakage, and 1550 cfm of filtered recirculation flow.

2.8.5 Radiological Consequences

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. 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 SG are assumed to occur from the nearest feedwater line. Releases from the unaffected SG are assumed to occur from the closest ADV.

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 30 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 for the different modes of control room operation during this 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.

Reg. Guide 1.183 does not provide any direct guidance for the acceptance criteria for this event. However, the SRP states that the acceptance criteria is "a small fraction" of the 10CFR100 values which is further described as 10% of the limit. In applying the AST methodology to the feedwater line break, the same 10% interpretation is applied to the 10CFR part 50.67 limits for the LPZ and EAB dose. The acceptable dose

limit for the Control Room (CR) is that specified in 10CFR50.67. For a Feedwater Line Break, these limits are interpreted as:

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

The radiological consequences of the Feedwater Line Break event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 2.8-4, "FWLB Dose Consequences," the radiological consequences of the Feedwater Line Break event are all within the appropriate acceptance criteria.

2.9 Environmental Qualification (EQ)

The St. Lucie Unit No. 2 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. 2 EQ analyses will continue to be based on TID-14844 assumptions.

3.0 Summary of Results

Results of the St. Lucie Unit 2 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, Letdown Line Break, and Feedwater Line Break (FWLB) have been made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the St. Lucie Unit No. 2 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 435 cfm.

5.0 References

5.1 St. Lucie Unit No. 2 Updated Final Safety Analysis Report (UFSAR), (through Amendment 17).

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. 2 Technical Specifications (through Amendment 146).

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, (RSISS Computer Code Collection No. CCC-445).

5.15 Numerical Applications Inc., NAI-9912-04, Revision 2, "RADTRAD-NAI Version 1.1a(QA) Documentation," October 2004.

5.16 PSL-ENG-SENS-02-058, "Engineering Evaluation, Alternate Source Term Design Inputs, St. Lucie Nuclear Plant, Unit 2," Revision 4, Florida Power & Light Company.

5.17 Florida Power & Light Company, St. Lucie Unit No. 2 Core Operating Limits Report (COLR).

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

5.19 USNRC Regulatory Guide 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. Report (UFSAR), (through Amendment 17).

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

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

5.24 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.

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

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

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

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

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

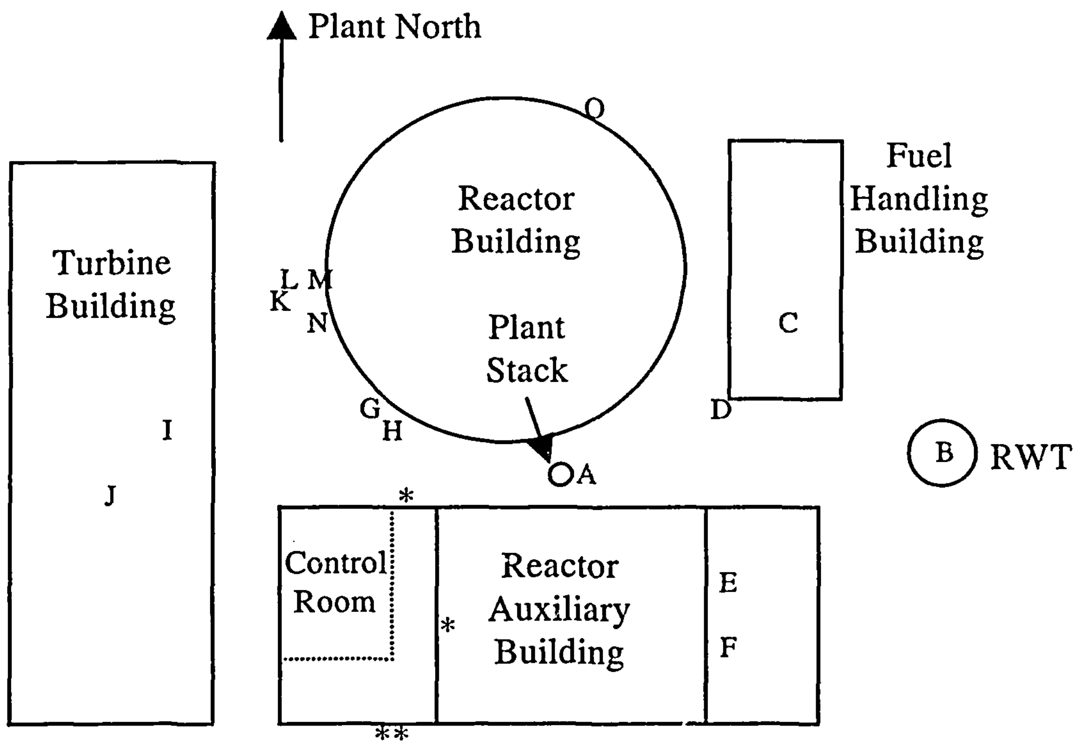
5.30 Kewaunee Nuclear Plant – Issuance of Amendment Regarding Implementation of Alternate Source Term (TAC No. MB4596), March 17, 2003.

5.31 Shearon Harris Nuclear Power Plant, Unit1 – Issuance of Amendment Re: Steam Generator Replacement and Power Uprate (TAC Nos. MB0199 and MB0782).

5.32 USNRC, Regulatory Issue Summary 2006-04, Experience With Implementation of Alternate Source Terms, March 7, 2006.

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 Stack
D – FHB Closest Point
E – Louver 2L-7B
F – Louver 2L-7A
G – Containment Personnel Lock
H – Containment Bulkhead Door
I – Steam Jet Air Ejector
J – Turbine Condenser
K – Closest ADV
L – Closest MSSV
M – Closest Main Steam Line Point (containment penetration)
N – Closest Feedwater Line Point (containment penetration)
O – Containment Maintenance Hatch

Table 1.6.3-1

Control Room Ventilation System Parameters

Parameter	Value
Control Room Volume	97,215 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	1000 cfm
Unfiltered Inleakage All events except MSLB MSLB	 500 cfm 435 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 All events except MSLB MSLB	 500 cfm 435 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 All events except MSLB MSLB	 500 cfm 435 cfm
Filter Efficiencies	
Particulate	99%
Elemental	99%
Organic	99%

Table 1.6.3-2

LOCA Direct Shine Dose

Source	Direct Shine Dose (rem)
Containment	0.03
Filters	0.18
External Cloud	0.07
Total	**0.28**

Table 1.7.2-1
Primary Coolant Source Term

Nuclide	µCi/gm	Nuclide	µCi/gm
I-131	0.8133	SR-90	2.428E-04
I-132	0.1692	CR-51	3.323E-03
I-133	1.0111	FE-59	1.789E-03
I-134	0.1011	CO-60	3.578E-03
I-135	0.5055	SR-91	4.729E-03
H-3	4.473E+00	Y-90	2.428E-04
KR-85M	1.406E+00	Y-91	7.796E-03
KR-85	2.939E+00	ZR-95	9.841E-03
KR-87	1.137E+00	MO-99	5.879E-01
KR-88	3.451E+00	RU-103	8.179E-03
RB-88	3.451E+00	RU-106	2.173E-03
RB-89	8.307E-02	TE-129	1.661E-02
XE-131M	5.623E+00	TE-132	4.473E-01
XE-133	3.706E+02	TE-134	4.217E-02
XE-135	8.179E+00	BA-140	1.125E-02
BR-84	3.706E-02	LA-140	1.086E-02
CS-134	2.045E-01	CE-144	5.879E-03
CS-136	1.406E-01	PR-143	9.329E-03
CS-137	5.623E-01	MN-54	5.495E-04
CS-138	1.176E+00	FE-55	2.812E-03
SR-89	7.285E-03	CO-58	2.812E-02

Table 1.7.3-1
Secondary Side Source Term

Isotope	μCi/gm
I-131	0.08133
I-132	0.01692
I-133	0.10111
I-134	0.01011
I-135	0.05055

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

Nuclide	Curies	Nuclide	Curies
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
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 1.8.1-1
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	59.67	18.2	48.07	14.6	58
Stack/Plant Vent	S CR intake	184	56.1	57.58	17.6	126.7	38.6	1
RWT	N CR intake	48.22	14.6	59.67	18.2	245.3	74.7	65
RWT	S CR intake	48.22	14.6	57.58	17.6	275.6	84.0	42
FHB Closest Point	N CR intake	43	13.1	59.67	18.2	121.3	36.9	48
FHB Closest Point	S CR intake	43	13.1	57.58	17.6	189.8	57.8	16
Aux. Bldg. Louver 2L-7B	N CR intake	38	11.5	59.67	18.2	123.81	37.7	72
Aux. Bldg. Louver 2L-7A	S CR intake	38	11.5	57.58	17.6	147.85	45.0	40
Condenser	N CR intake	5.25	1.6	59.67	18.2	153.23	46.7	244
Closest ADV	N CR intake	47	14.3	59.67	18.2	100.66	30.6	298
Closest ADV	S CR intake	47	14.3	57.58	17.6	195.55	59.6	320
Closest Feedwater Line Point	N CR intake	17	5.2	59.67	18.2	83.06	25.3	306
Closest Feedwater Line Point	S CR intake	17	5.2	57.58	17.6	183.25	55.8	325
Containment Maintenance Hatch	N CR intake	16	4.9	59.67	18.2	172.17	52.4	359
Containment Maintenance Hatch	S CR intake	16	4.9	57.58	17.6	276.7	84.3	351
Stack/Plant Vent	Midpoint between intakes	184	56.1	58.625	17.9	72.02	21.9	10
RWT	Midpoint between intakes	48.22	14.6	58.625	17.9	244.78	74.6	53

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
Aux. Bldg. Louver 2L-7A	Midpoint between intakes	38	11.5	58.625	17.9	119	36.2	61
Closest ADV	Midpoint between intakes	47	14.3	58.625	17.9	149.54	45.5	309
Closest Feedwater Line Point	Midpoint between intakes	17	5.2	58.625	17.9	134.2	40.9	315
Containment Maintenance Hatch	Midpoint between intakes	16	4.9	58.625	17.9	220.1	67.0	352
FHB Closest Point	Midpoint between intakes	43	13.1	58.625	17.9	141.04	42.9	26
Stack/Plant Vent	Louver 2L-11	184	56.1	50.75	15.5	122.54	37.3	358
RWT	Louver 2L-11	48.22	14.6	50.75	15.5	267.44	81.5	42
Aux. Bldg. Louver 2L-7A	Louver 2L-11	38	11.5	50.75	15.5	139.81	42.6	38
Closest ADV	Louver 2L-11	47	14.3	50.75	15.5	197.4	60.1	317
Closest Feedwater Line Point	Louver 2L-11	17	5.2	50.75	15.5	184.33	56.1	322
Containment Maintenance Hatch	Louver 2L-11	16	4.9	50.75	15.5	273.79	83.4	350
FHB Closest Point	Louver 2L-11	43	13.1	50.75	15.5	183.73	56	14

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. For events where the limiting unfiltered inleakage location is through the control room intakes, atmospheric dispersion factors corresponding to the midpoint between the control room intakes are to be used during the time period when the control room intakes are isolated.

7. The containment/shield building penetrations at the 228° and 240° azimuths empty into the covered walkway that leads to the RAB; therefore, 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 *X/Q* factors for the control room intakes and for switchgear room louver 2L-11 that apply to the various accident scenarios. For the intakes, values are presented for the unfavorable intake prior to control room isolation, the midpoint between the intakes during isolation, as well as values for the favorable intake following manual restoration of filtered control room make-up flow. These values are not corrected for Control Room Occupancy Factors but do include 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.

The atmospheric dispersion factors corresponding to ADVs were determined to be more limiting than those from the MSSVs for all time periods. Therefore, the more limiting ADV values have been used throughout the analyses for all secondary releases. No distinction is made between automatic steam relief from the MSSVs and controlled releases from the ADVs for radiological purposes.

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*	6.48E-04	4.28E-04	1.99E-04	1.20E-04	9.15E-05
C	RWT	N CR intake	1.38E-03				
D	RWT	S CR intake	1.01E-03	8.64E-04	3.72E-04	2.92E-04	2.20E-04
E	FHB Closest Point	N CR intake	4.86E-03				
F	FHB Closest Point	S CR intake	1.86E-03	1.37E-03	6.14E-04	3.90E-04	3.05E-04
G	Aux. Bldg. Louver 2L-7B	N CR intake	4.85E-03				
H	Aux. Bldg. Louver 2L-7A	S CR intake	3.11E-03	2.73E-03	1.17E-03	8.73E-04	6.76E-04
I	Condenser	N CR intake	2.47E-03				
J	Closest ADV #	N CR intake	6.69E-03				
K	Closest ADV #	S CR intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-04
L	Closest Feedwater Line Point	N CR intake	7.30E-03				
M	Closest Feedwater Line Point	S CR intake	1.94E-03	1.50E-03	6.47E-04	4.32E-04	3.22E-04

Release-Receptor Pair	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
N	Stack/Plant Vent	Midpoint between intakes	3.79E-03				
O	RWT	Midpoint between intakes	1.33E-03				
P	Aux. Bldg. Louver L-7A	Midpoint between intakes	5.04E-03				
Q	Closest ADV #	Midpoint between intakes	3.11E-03				
R	Closest Feedwater Line Point	Midpoint between intakes	3.32E-03				
S	Containment Maintenance Hatch	N CR intake	1.87E-03				
T	Containment Maintenance Hatch	S CR intake	8.17E-04	6.08E-04	2.84E-04	1.71E-04	1.29E-04
U	Containment Maintenance Hatch	Midpoint between intakes	1.24E-03				
V	FHB Closest Point	Midpoint between intakes	3.27E-03				
W	Stack/Plant Vent	Louver 2L-11	2.55E-03	1.71E-03	7.99E-04	4.73E-04	3.56E-04
X	RWT	Louver 2L-11	1.08E-03	9.06E-04	3.95E-04	3.12E-04	2.31E-04
Y	Aux. Bldg. Louver L-7A	Louver 2L-11	3.58E-03	2.98E-03	1.28E-03	9.38E-04	7.27E-04
Z	Closest ADV #	Louver 2L-11	1.86E-03	1.44E-03	5.78E-04	4.15E-04	3.14E-04
AA	Closest Feedwater Line Point	Louver 2L-11	1.95E-03	1.50E-03	6.42E-04	4.40E-04	3.24E-04
BB	Containment Maintenance Hatch	Louver 2L-11	9.40E-04	6.98E-04	3.28E-04	1.97E-04	1.45E-04
CC	FHB Closest Point	Louver 2L-11	2.01E-03	1.43E-03	6.48E-04	4.19E-04	3.17E-04

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

Event	Prior to Control Room Isolation	During Control Room Isolation	After Initiation of Filtered Air Make-up
LOCA:			
- Containment Leakage (SBVS)	A	N	B
- Containment (SBVS Bypass)	L	R	M
- ECCS Leakage	G	P	H
- RWT Backleakage	C	O	D
- Cont. Purge/H_2 Purge	A	N	B
FHA:			
Containment Release	S	U	T
FHB Release	E	V	F
MSLB:			
- Outside Containment.	J	Q	K
- Inside Containment (SBVS)	A	N	B
- Inside Containment (SBVS Bypass)	L	R	M
SGTR	I (Prior to Turbine Trip) J (After Turbine Trip)	Q	K
Locked Rotor	J	Q	K
CEA Ejection:			
- Secondary Release	J	Q	K
- Inside Containment (SBVS)	A	N	B
- Inside Containment (SBVS Bypass)	L	R	M
Letdown Line Break:			
- RCS Release	A	N	B
- SG Release	J	Q	K
FWLB:			
- Intact SG	J	Q	K
- Affected SG	L	R	M

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	1.10E-04	1.06E-04
0-8 hours	6.17E-05	5.91E-05
8-24 hours	4.61E-05	4.41E-05
1-4 days	2.45E-05	2.33E-05
4-30 days	9.93E-06	9.32E-06

The above table summarizes the maximum *X/Q* factors for the EAB and LPZ. Note that the 0-2 hour EAB *X/Q* factor was used for the entire event.

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)
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 and 3.2
<u>ECCS Systems Leakage</u>	
Sump Volume (minimum)	55,739 ft^3
ECCS Leakage to RAB (2 times allowed value)	1.28 gph
Flashing Fraction	Calculated – 3.4% Used for dose determination – 10%
Chemical form of the iodine in the sump water	0% aerosol, 97% elemental, and 3.0% organic
Release ECCS Area Filtration Efficiency	Elemental – 95% Organic – 95% Particulate – 99% (100% of the particulates are retained in the ECCS fluid)

Input/Assumption	Value
RWT Back-leakage	
Sump Volume (at time of recirculation)	58,894 ft^3
ECCS Leakage to RWT (2 times allowed value)	2 gpm
Flashing Fraction (elemental Iodine assumed to be released into tank space based upon partition factor)	0 % based on temperature of fluid reaching RWT
RWT liquid/vapor Elemental Iodine partition factor	Table 2.1-7
Elemental Iodine fraction in RWT	Table 2.1-6
Initial RWT Liquid Inventory (minimum)	52,345 gal
Release from RWT Vapor Space	Table 2.1-8
Containment Purge Release	2500 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	2,125,000 ft^3
Containment Unsprayed Region Volume	375,000 ft^3
Flowrate between sprayed and unsprayed volumes	12,500 cfm
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Particulate Iodine Time to reach DF of 50	 20/hour 3.06 hours 6.40/hour 2.65 hours
Spray Initiation Time Spray Termination Time	60 seconds (0.01667 hours) 8 hours
Control Room Ventilation System Time of automatic control room isolation Time of manual control room unisolation	Table 1.6.3-1 30 seconds 1.5 hrs
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 95% Organic – 95%
Secondary Containment Drawdown Time	310 seconds
Secondary Containment Bypass Fraction	9.6%
Containment Purge Filtration	0 %

Input/Assumption	Value
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	Nearest containment penetration to CR ventilation intake
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 LOCA Time Dependent RWT pH

Time (hours)	SIRWT pH
0.00	4.900
0.33	4.900
0.50	4.900
0.64	4.900
0.83	4.900
2.78	4.902
4.17	4.904
5.56	4.905
9.72	4.909
11.11	4.911
15.28	4.915
22.22	4.921
55.56	4.951
83.33	4.975
97.22	4.987
111.11	4.998
152.78	5.029
194.44	5.059
250.00	5.095
305.56	5.128
402.78	5.181
500.00	5.228
597.22	5.271
694.44	5.309
720.00	5.319

Table 2.1-4
LOCA Time Dependent RWT Total Iodine Concentration*

Time (hours)	RWT Iodine Concentration (gm-atom/liter)
0.00	0.00E+00
0.33	0.00E+00
0.50	2.27E-08
0.64	4.17E-08
0.83	6.75E-08
2.78	3.31E-07
4.17	5.18E-07
5.56	7.03E-07
9.72	1.25E-06
11.11	1.43E-06
15.28	1.97E-06
22.22	2.84E-06
55.56	6.67E-06
83.33	9.49E-06
97.22	1.08E-05
111.11	1.20E-05
152.78	1.54E-05
194.44	1.83E-05
250.00	2.16E-05
305.56	2.44E-05
402.78	2.85E-05
500.00	3.17E-05
597.22	3.43E-05
694.44	3.65E-05
720.00	3.70E-05

*Includes radioactive and stable iodine isotopes

Table 2.1-5 LOCA
Time Dependent RWT Liquid Temperature

Time (hr)	Temperature (°F)
0.00	100.0
0.33	100.0
0.50	100.0
0.64	100.0
0.83	100.0
2.78	100.0
4.17	100.0
5.56	100.0
9.72	100.0
11.11	100.0
15.28	100.0
22.22	100.0
55.56	101.3
83.33	102.2
97.22	102.5
111.11	102.8
152.78	103.3
194.44	103.6
250.00	103.8
305.56	103.8
402.78	103.9
500.00	103.9
597.22	103.8
694.44	103.8
720.00	103.8

Table 2.1-6
LOCA Time Dependent RWT Elemental Iodine Fraction

Time (hr)	Elemental Iodine Fraction
0.00	0.00E+00
0.33	0.00E+00
0.50	1.18E-04
0.64	2.18E-04
0.83	3.52E-04
2.78	1.71E-03
4.17	2.65E-03
5.56	3.57E-03
9.72	6.22E-03
11.11	7.07E-03
15.28	9.52E-03
22.22	1.33E-02
55.56	2.70E-02
83.33	3.44E-02
97.22	3.71E-02
111.11	3.94E-02
152.78	4.39E-02
194.44	4.62E-02
250.00	4.69E-02
305.56	4.63E-02
402.78	4.36E-02
500.00	4.01E-02
597.22	3.66E-02
694.44	3.32E-02
720.00	3.24E-02

Table 2.1-7
LOCA Time Dependent RWT Partition Coefficient

Time (hr)	Elemental Iodine Partition Coefficient
0.00	45.65
0.33	45.65
0.50	45.65
0.64	45.65
0.83	45.65
2.78	45.65
4.17	45.65
5.56	45.65
9.72	45.65
11.11	45.65
15.28	45.65
22.22	45.65
55.56	44.53
83.33	43.77
97.22	43.52
111.11	43.27
152.78	42.86
194.44	42.62
250.00	42.46
305.56	42.46
402.78	42.38
500.00	42.38
597.22	42.46
694.44	42.46
720.00	42.46

Table 2.1-8 LOCA Release Rate from RWT

Time (hours)	Adjusted Iodine Release Rate (cfm)
0.33	2.637E-07
4.17	1.165E-06
11.11	3.512E-06
22.22	2.847E-05
111.11	1.110E-04
305.56	1.759E-04
402.78	1.915E-04
500.00	1.995E-04
597.22	2.033E-04
694.44	1.867E-04

Table 2.1-9 LOCA Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	1.16	2.56	4.46
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 – 285 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
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 30 seconds 1.5 hours 500 cfm
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)
Containment Release	0.29	0.28	0.81
FHB Release	0.29	0.28	1.63
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 for MSLB Outside of Containment	1.8%
% DNB for MSLB Inside of Containment	29%
% Fuel Centerline Melt for MSLB Outside of Containment	0.43%
% Fuel Centerline Melt for MSLB Inside of Containment	6.1%
Core Fission Product Inventory	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 Fraction from DNB Fuel Failures	RG 1.183, Section 3.2
Release Fraction from Centerline Melt Fuel Failures	RG 1.183, Section 3.2, and Section 1 of Appendix H
Steam Generator Tube Leakage	0.25 gpm per SG (Table 2.3-3)
Time to Terminate SG Tube Leakage	12 hours
Steam Release from Intact SGs	Table 2.3-2
Intact SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	 1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Unaffected SG - 100 Faulted SG - None
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.50E+06 ft^3 0.5% (by weight)/day 0.25% (by weight)/day
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 95% Organic – 95%
Secondary Containment Drawdown Time	310 seconds
Secondary Containment Bypass Fraction	9.6%
RCS Mass	423,700 lb_m Minimum mass used for fuel failure dose contribution to maximize SG tube leakage activity.

Input/Assumption	Value
SG Secondary Side Mass	minimum – 105,000 lb_m (per SG) maximum – 260,000 lb_m (per 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%
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 Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 30 seconds 1.5 hours 435 cfm
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	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.0	5600
1.0 – 1.5	5250
1.5 – 2.25	4934
2.25 – 4.0	4098
4.0 – 8.0	2671
8.0 – 10.32	3247
10.32 - 720	0

Table 2.3-3 MSLB Steam Generator Tube Leakage

Time (hours)	Tube Leakage per SG (lb_m/min)
0 – 0.50	1.47
0.50 – 1.0	1.52
1.0 – 1.5	1.62
1.5 – 2.0	1.71
2.0 – 2.5	1.78
2.5 – 3.0	1.85
3.0 – 3.5	1.90
3.5 – 9.69	1.92
9.69 – 12	1.96
12 - 720	0

Table 2.3-4 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	1.8% DNB	0.33	0.91	4.69
MSLB – Outside of Containment	0.43% FCM	0.37	0.98	4.77
MSLB – Inside of Containment	29% DNB	0.54	1.09	4.96
MSLB – Inside of Containment	6.1% FCM	0.79	1.50	4.92
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
Iodine Spike Appearance Rate	335 times
Duration of Accident-Initiated Spike	8 hours
Integrated Break Flow and Steam Release	Table 2.4-2
Break Flow Flashing Fraction	Prior to Reactor Trip - 17% Following Reactor Trip – 5%
Time to Terminate Break Flow	30 minutes
Steam Generator Tube Leakage Rate	0.25 gpm per SG
Time to Terminate Tube Leakage	12 hours
Time to Re-cover Intact SG Tubes	1 hour
Steam Generator Secondary Side Partition Coefficients	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
RCS Mass	Pre-accident Iodine spike – 423,700 lb_m Concurrent Iodine spike – 452,000 lb_m
SG Secondary Side Mass	minimum – 105,000 lb_m (per SG) maximum – 260,000 lb_m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
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 Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 409.2 seconds 1.5 hours 500 cfm
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	78,040	0 – 0.1053 hrs : 661,842 (via Condenser) 0.1053 – 0.5 hrs: 88,352 (via MSSV)	656,568 (via Condenser) 86,821 (via MSSVs)
0.5 – 2.0	0	0	601,096 (via ADVs)
2 – 8	N/A	N/A	876,233
8 – 10.32	N/A	N/A	32.47 lb_m/min

[1] Flowrate assumed to be constant within time period

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

Isotope	Activity (μCi/gm)
Iodine-131	48.8
Iodine-132	10.2
Iodine-133	60.7
Iodine-134	6.07
Iodine-135	30.3

Table 2.4-4 SGTR Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	150 gpm at 120°F, 650 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	452,000 lb_m
I-131 Decay Constant	0.002951
I-132 Decay Constant	0.007914
I-133 Decay Constant	0.003446
I-134 Decay Constant	0.016069
I-135 Decay Constant	0.004639

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

Isotope	Activity Appearance Rate (Ci/min)	Total 8-hour Production (Ci)
Iodine-131	164.8	79124
Iodine-132	92.0	44146
Iodine-133	239.3	114868
Iodine-134	111.6	53559
Iodine-135	161.1	77310

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	0.25	0.24	2.57
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	0.06	0.06	0.66
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 Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 4.5w/o
Maximum Radial Peaking Factor	1.7
Percent of Fuel Rods in DNB	13.7%
Core Fission Product Inventory	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 Fraction from Breached Fuel	RG 1.183, Section 3.2
Steam Generator Tube Leakage	0.5 gpm (Table 2.5-3)
Time to Terminate SG Tube Leakage	12 hours
Secondary Side Mass Releases to Environment	Table 2.5-2
SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	 1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
RCS Mass	423,700 lb_m Minimum mass used for fuel failure dose contribution to maximize SG tube leakage activity.
SG Secondary Side Mass	minimum – 105,000 lb_m (per SG) maximum – 260,000 lb_m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
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 Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 30 seconds 1.5 hours 500 cfm
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)	SG Steam Release Rate (lb_m/min)
0 – 0.25	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.0	5600
1.0 – 1.5	5250
1.5 – 2.25	4934
2.25 – 4.0	4098
4.0 – 8.0	2671
8.0 – 10.32	3247

Table 2.5-3 Locked Rotor Steam Generator Tube Leakage

Time (hours)	SG Tube Leakage (lb_m/min)
0 – 0.50	2.94
0.50 – 1.0	3.05
1.0 – 1.5	3.25
1.5 – 2.0	3.42
2.0 – 2.5	3.57
2.5 – 3.0	3.70
3.0 – 3.5	3.80
3.5 – 9.69	3.83
9.69 – 12	3.91
12 - 720	0

Table 2.5-4 Locked Rotor Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Locked Rotor	0.25	0.56	2.81
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
Percent of Fuel Rods in DNB	9.5%
Percent of Fuel Rods with Centerline Melt	0.5%
Core Fission Product Inventory	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 Fraction from DNB Fuel Failures	Section 1 of Appendix H to RG 1.183
Release Fraction from Centerline Melt Fuel Failures	Section 1 of Appendix H to RG 1.183
Steam Generator Tube Leakage	0.5 gpm (Table 2.6-3)
Time to Terminate SG Tube Leakage	12 hours
Secondary Side Mass Releases to Environment	Table 2.6-2
SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	 1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
RCS Mass	minimum – 423,700 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 (per SG) maximum – 260,000 lb_m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
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	 Table 1.8.2-1

Input/Assumption	Value
Onsite	Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 30 seconds 1.5 hours 500 cfm
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 Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.50E+06 ft^3 0.5% (by weight)/day 0.25% (by weight)/day
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 95% Organic – 95%
Secondary Containment Drawdown Time	310 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	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.0	5600
1.0 – 1.5	5250
1.5 – 2.25	4934
2.25 – 4.0	4098
4.0 – 8.0	2671
8.0 – 10.32	3247

Table 2.6-3 CEA Ejection Steam Generator Tube Leakage

Time (hours)	SG Tube Leakage (lb_m/min)
0 – 0.50	2.94
0.50 – 1.0	3.05
1.0 – 1.5	3.25
1.5 – 2.0	3.42
2.0 – 2.5	3.57
2.5 – 3.0	3.70
3.0 – 3.5	3.80
3.5 – 9.69	3.83
9.69 – 12	3.91
12 - 720	0

Table 2.6-4 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	0.26	0.52	2.78
CEA Ejection – Secondary Release	0.30	0.65	2.87
Acceptance Criteria	6.3	6.3	5

[1] Worst 2-hour dose
[2] Integrated 30-day dose

Table 2.7-1
Letdown Line Rupture – 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)
Iodine spike Appearance Rate	500 times
Duration of Accident Initiated Spike	8 hrs
Steam Generator Tube Leakage	0.5 gpm (Table 2.7-5)
Time to Terminate SG Tube Leakage	12 hours
Secondary Side Mass Releases to Environment	Table 2.7-2
SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	 1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
RCS mass	RCS Activity – 423,700 lb_m Concurrent Iodine spike – 452,000 lb_m
SG Secondary Side Mass	minimum – 105,000 lb_m (per SG) maximum – 260,000 lb_m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
Letdown Line Rupture flow rate	85,788 lb_m over 1920 seconds
Letdown Line Flashing Fraction	25.9%
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 30 seconds 1.5 hours 500 cfm
Chemical Form of Iodine Released	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
Breathing rates Offsite Control Room	RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

Table 2.7-2 Letdown Line Rupture Steam Release Rate

Time (hours)	SG Steam Release Rate (lb_m/min)
0 – 0.25	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.0	5600
1.0 – 1.5	5250
1.5 – 2.25	4934
2.25 – 4.0	4098
4.0 – 8.0	2671
8.0 – 10.32	3247

Table 2.7-3 Letdown Line Rupture Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	150 gpm at 120°F, 650 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	452,000 lb_m
I-131 Decay Constant	0.002951
I-132 Decay Constant	0.007914
I-133 Decay Constant	0.003446
I-134 Decay Constant	0.016069
I-135 Decay Constant	0.004639

Table 2.7-4 Letdown Line Rupture
Concurrent Iodine Spike (500 x) Activity Appearance Rate

Isotope	Activity Appearance Rate (Ci/min)	Total 8-hour Production (Ci)
Iodine-131	0.4920	118077
Iodine-132	0.2745	65880
Iodine-133	0.7144	171445
Iodine-134	0.3330	79928
Iodine-135	0.4807	115368

Table 2.7-5 Letdown Line Rupture Steam Generator Tube Leakage

Time (hours)	SG Tube Leakage (lb_m/min)
0 – 0.50	2.94
0.50 – 1.0	3.05
1.0 – 1.5	3.25
1.5 – 2.0	3.42
2.0 – 2.5	3.57
2.5 – 3.0	3.70
3.0 – 3.5	3.80
3.5 – 9.69	3.83
9.69 – 12	3.91
12 - 720	0

Table 2.7-6 Letdown Line Rupture Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Letdown Line Rupture	0.36	0.36	2.57
Acceptance Criteria	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 2.8-1
Feedwater Line Break (FWLB)– 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)
Steam Generator Tube Leakage	0.25 gpm per SG (Table 2.8-3)
Time to Terminate SG Tube Leakage	12 hours
Unaffected Secondary Side Mass Releases to Environment	Table 2.8-2
Intact SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	 1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
Unaffected SG Secondary Side Mass	minimum – 105,000 lb_m (one SG) 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%
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 Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	 30 seconds 1.5 hours 500 cfm
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 FWLB Steam Release Rate

Time (hours)	Unaffected SG Steam Release Rate (lb_m/min)
0 – 0.25	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.0	5600
1.0 – 1.5	5250
1.5 – 2.25	4934
2.25 – 4.0	4098
4.0 – 8.0	2671
8.0 – 10.32	3247

Table 2.8-3 FWLB Steam Generator Tube Leakage

Time (hours)	Unaffected SG Tube Leakage (lb_m/min)
0 – 0.50	1.47
0.50 – 1.0	1.52
1.0 – 1.5	1.62
1.5 – 2.0	1.71
2.0 – 2.5	1.78
2.5 – 3.0	1.85
3.0 – 3.5	1.90
3.5 – 9.69	1.92
9.69 – 12	1.96
12 - 720	0

Table 2.8-4 FWLB Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
FWLB	0.02	0.02	0.82
Acceptance Criteria	2.5 (3)	2.5 (3)	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) "Well within" is taken as 10% of the 25 rem TEDE limit from 10CFR50.67

Table 3-1
St. Lucie Plant, Unit No. 2
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	500	1.16	2.56	4.46
MSLB – Outside of Containment (1.8% DNB)	435	0.33	0.91	4.69
MSLB – Outside of Containment (0.43% FCM)	435	0.37	0.98	4.77
MSLB – Inside of Containment (29% DNB)	435	0.54	1.09	4.96
MSLB – Inside of Containment (6.1% FCM)	435	0.79	1.50	4.92
SGTR Pre-accident Iodine Spike	500	0.25	0.24	2.57
Acceptance Criteria		**≤ 25(3)**	**≤ 25(3)**	**≤ 5(4)**
SGTR Concurrent Iodine Spike	500	0.06	0.06	0.66
Locked Rotor (13.7% DNB)	500	0.25	0.56	2.81
FWLB*	500	0.02	0.02	0.82
Letdown Line Rupture*	500	0.36	0.36	2.57
Acceptance Criteria		**≤ 2.5 (3)**	**≤ 2.5 (3)**	**≤ 5 (4)**
FHA - Containment	500	0.29	0.28	0.81
FHA – Fuel Handling Building	500	0.29	0.28	1.63
CEA Ejection – Containment Release (9.5% DNB, 0.5% FCM)	500	0.26	0.52	2.78
CEA Ejection – Secondary Release (9.5% DNB, 0.5% FCM)	500	0.30	0.65	2.87
Acceptance Criteria		**≤ 6.3(3)**	**≤ 6.3(3)**	**≤ 5(3)**

(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