UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 26, 2008

Mr. J. A. Stall
Senior Vice President, Nuclear and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: ST. LUCIE PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM (TAC NO. MD6173)

Dear Mr. Stall:

The Commission has issued the enclosed Amendment No. 206 to Renewed Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No.1. This amendment consist of changes to the Technical Specifications in response to your application dated July 16, 2007, as supplemented by letters dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008.

This amendment modifies the facility's operating licensing bases to adopt the alternative source term as allowed in 10 CFR 50.67 and described in Regulatory Guide 1.183. The licensee revised the plant licensing basis through reanalysis of the following radiological consequences of the Updated Final Safety Analysis Report Chapter 15 accidents: Loss-of-Coolant Accident, Fuel-Handling Accident, Main Steam Line Break, Steam Generator Tube Rupture, Reactor Coolant Pump Shaft Seizure, Control Element Assembly Ejection, and Inadvertent Opening of a Main Steam Safety Valve.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Brenda Mozafari

Brenda L. Mozafari, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosures:
1. Amendment No. 206 to DPR-67
2. Safety Evaluation

cc w/enclosures: Distribution via ListServ

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 206
Renewed License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company (the licensee), dated July 16, 2007, as supplemented by letters dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Renewed Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 9 months.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: November 26, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 206

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace Page 3 of Renewed Operating License DPR-67 with the attached page 3.

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
Page 1-3	Page 1-3
Page 3/4 3-22	Page 3/4 3-22a
Page 3/4 3-23	Page 3/4 3-23
Page 3/4 3-24	Page 3/4 3-24
Page 3/4 3-25	Page 3/4 3-25
Page 3/4 6-27	Page 3/4 6-27
Page 3/4 6-28	Page 3/4 6-28
Page 3/4 6-29	Page 3/4 6-29
Page 3/4 7-24	Page 3/4 7-24
Page 3/4 7-25	Page 3/4 7/25
Page 3/4 7-26	Page 3/4 7-26
Page 6-15b	Page 6-15b
Page 6-15d	Page 6-15d
Page 6-15e	Page 6-15e

applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 28, 2003, describes certain future activities to be completed before the period of extended operation. FPL shall complete these activities no later than March 1, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on March 28, 2003, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed license. Until that update is complete, FPL may make changes to the programs described in such supplement without prior Commission approval, provided that FPL evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Sustained Core Uncovery Actions

Procedural guidance shall be in place to instruct operators to implement actions that are designed to mitigate a small-break loss-of-coolant accident prior to a calculated time of sustained core uncovery.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES REPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. 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.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

4.3.3.2 At lease once per 18 months, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area	1	*	≤ 15 mR/hr	$10^{-1} - 10^{4}$ mR/hr	13
b. Containment (CIS)	3	****	≤ 90 mR/hr	$1 - 10^{5}$ mR/hr	16
c. Containment Area – Hi Range	1	1, 2, 3, & 4	≤ 10 R/hr	$1 - 10^{7}$ R/hr	15
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	$10 - 10^{7}$ cpm	17
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14
b. Fuel Storage Pool Area Ventilation System					
i. Gaseous Activity	1	**	***	$10^{-7} - 10^{5}$ μCi/cc	12
ii. Particulate Activity	1	**	***	$1 - 10^{6}$ cpm	12

* With fuel in the storage pool or building.
** With recently irradiated fuel in the storage pool.
*** The Alarm Setpoints are determined and set in accordance with requirements of the Offsite Dose Calculation Manual.
**** During movement of recently irradiated fuel assemblies within containment.

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
c. Noble Gas Effluent Monitors					
i. Radwaste Building Exhaust System (Plant Vent Exhaust Monitor)	1	1, 2, 3 & 4	***	$10^{-7} - 10^{5}$ μCi/cc	15
ii. Steam Generator Blowdown Treatment Facility Building Exhaust System	1	1, 2, 3 & 4	***	$10^{-7} - 10^{-2}$ μCi/cc	15
iii. Steam Safety Valve Discharge	1/Header	1, 2, 3 & 4	***	$10^{-1} - 10^{3}$ μCi/cc	15
iv. ECCS Exhaust	1/Train	1, 2, 3 & 4	***	$10^{-7} - 10^{5}$ μCi/cc	15

*** The Alarm Setpoints are determined and set in accordance with requirements of the Offsite Dose Calculation Manual.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 12 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

2) Prepare and submit a Special Report to the Commission pursuant to Specification6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area	S	R	M	*
b. Containment (CIS)	S	R	M	***
c. Containment Area – High Range	S	R	M	1, 2, 3 & 4
d. Control Room Isolation	S	R	M	All Modes
2. PROCESS MONITORS				
a. Fuel Storage Pool Area – Ventilation System				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity RCS Leakage Detection	S	R	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building.
** With irradiated fuel in the storage pool.
*** During movement of recently irradiated fuel within containment.

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one shield building 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

4.6.6.1 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 filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

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

c. At least once per 18 months by:

   1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

   2. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

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

   4. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

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3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two independent ECCS area exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ECCS area exhaust air filter train inoperable, restore the inoperable train 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

4.7.8.1 Each ECCS area exhaust air filter train 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 filter and charcoal adsorber train and verifying that the train 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:

   1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

   2. Verifying that the filter train starts on a Safety Injection Actuation Signal.

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

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

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

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

h. Containment Leakage Rate Testing Program

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

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

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

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

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

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ 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 rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

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

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 Ventilation	≤ 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 Ventilation	≤ 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 Ventilation	≤ 2.5%	70%
Shield Building Ventilation System	≤ 2.5%	70%
ECCS Area Ventilation System	≤ 2.5%	70%

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 Ventilation	< 4.15" W.G.	2000 ± 200 cfm
Shield Building Ventilation System	≤ 6.15" W.G.	6000 ± 600 cfm
ECCS Area Ventilation System	< 4.15" 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

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator 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 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 gpm total through all SGs and 0.25 gpm through any one SG.

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

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

l. Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment of the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access Management System
ADV	atmospheric dump valve
AGL	above ground level
AEC	Atomic Energy Commission
ANSI	American National Standards Institute
ARC	alternate repair criteria
ASME	American Society of Mechanical Engineers
AST	alternative source term
cc/hour	cubic centimeters per hours
CDE	committed dose equivalent
CEA	control element assembly
CEDE	committed effective dose equivalent
CEDM	control element drive mechanism
cfm	cubic feet per minute
CFR	Code of Federal Regulations
CIAS	containment isolation actuation signal
CLB	current licensing basis
CR	control room
CRACS	control room air conditioning system
CRE	control room envelope
CREVS	control room emergency ventilation system
CROAI	control room outside air isolation
CsI	cesium iodide
DBA	design-basis accident

DCF	dose conversion factor
DEI	dose equivalent I-131
DF	decontamination factor
DNB	departure from nucleate boiling
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EDG	emergency diesel generator
ESF	engineered safety feature
EQ	equipment qualification
°F	degrees Fahrenheit
FCM	fuel centerline melt
FGR	Federal Guidance Report
FHA	fuel-handing accident
FHB	fuel-handing building
FWLB	feedwater line break
GDC	General Design Criteria / Criterion
gpd	gallons per day
gph	gallons per hour
gpm	gallons per minute
HEPA	high-efficiency particulate air
IOMSSV	inadvertent opening of a main steam safety valve
JFD	joint frequency distribution
kw/ft	kilowatt per foot
LAR	license amendment request
LCO	limiting condition for operation
LOCA	loss-of-coolant accident

LOOP	loss of offsite power
LPZ	low-population zone
LRA	locked rotor accident
LWR	light-water reactor
MS	main stack/plant vent
m/s	meters per second
MSLB	main steam line break
MSSV	main steam safety valve
MWt	megawatts thermal
MWD/MTU	megawatt days per metric ton of uranium
µCi/gm	micro curie per gram
NAI	Numerical Applications, Inc.
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PC	partition coefficient
PORV	power operated relief valve
PWR	pressurized-water reactor
RAB	Reactor Auxiliary Building
RCP	reactor coolant pump
RCS	reactor coolant system
rem	roentgen equivalent man
RG	Regulatory Guide
RIS	Regulatory Issue Summary
RWT	refueling water tank
SBVS	shield building ventilation system
SDC	shutdown cooling
SE	safety evaluation

SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
SIS	safety injection system
SR	surveillance requirement
SRP	Standard Review Plan
TEDE	total effective dose equivalent(
TID	Technical Information Document
TSC	technical support center
TSTF	Technical Specification Task Force Traveler
UFSAR	Updated Final Safety Analysis Report
VFTP	Ventilation System to the Ventilation Filter Testing Program
w/o	weight percent

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated July 16, 2007, as supplemented by letters dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008, Florida Power and Light Company (the licensee) requested to amend Renewed Operating License DPR-67 for St. Lucie Unit 1, in order to fully implement an alternative source term (AST) methodology. The application provides the technical specification (TS) changes and evaluations of the radiological consequences of design-basis accidents (DBAs) for implementation of a full-scope AST pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 and using the methodology described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors."

The licensee's supplements dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008, provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the *Federal Register* and did not change the initial proposed no significant hazards determination.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff reviewed the licensee's evaluation of the radiological consequences of affected DBAs for implementation of the AST methodology, and the associated changes to the TS proposed by the licensee, against the requirements specified in 10 CFR 50.67(b)(2). It states in 10 CFR 50.67(b)(2) that the licensee's analysis demonstrates with reasonable assurance that:

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

- An individual located at any point on the outer boundary of the low-population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission

product release during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE.

- Adequate radiation protection is provided to permit access to and occupancy of the control room (CR) under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, and the accident specific guideline values in Regulatory Position 4.4 of RG 1.183 and Table 1 of Standard Review Plan (SRP) Section 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulations, regulatory guides, and standards:

- 10 CFR Part 50.67, "Accident Source Term."

- 10 CFR Part 50, Appendix A, "General Design Criterion (GDC) for Nuclear Power Plants": GDC 19, "Control room."

- RG 1.23, "Onsite Meteorological Programs," February 1972.

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Rev.1, March 2007.

- RG 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.

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, November 1982.

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000.

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Rev. 0, June 2003.

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Rev. 0, May 2003.

- NUREG-0409, "Iodine Behavior in a PWR [Pressurized-Water Reactor] Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985.

- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Rev. 3, March 2007.

- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Rev. 3, March 2007.

- NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Rev. 4, March 2007.

- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000.

- NUREG-0800, "Standard Review Plan," Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Rev. 2, July 1981.

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

The construction permit issued to St. Lucie Unit 1 on July 1, 1970, preceded the publication of the Atomic Energy Commission (AEC) GDC, as well as the NUREG-0800 SRPs. Prior to the implementation of AST, there was no automatic isolation feature for CR Outside Air Isolation (CROAI) at St. Lucie Unit 1. The CR isolation on high-radiation was based on operator action after receipt of a high radiation alarm. In addition, normally only one radiation monitor was placed in operation. AST analysis requires that the CR ventilation system be single failure proof and should automatically isolate from the outside air on sensing high radiation at the outside air intake and place the ventilation system in the recirculation mode. The existing design could not support CROAI on failure of the only operating radiation monitor.

In the August 13, 2008, letter, the licensee described the following modifications to achieve automatic ventilation system isolation, and ensure the ventilation system is in the recirculation mode on sensing high radiation in the outside air intake.

a) Redundant safety-related seismically qualified radiation detectors will be provided in both the north and the south CROAI ducts.

b) The radiation detectors will be seismically mounted and are not susceptible to flood or missile threat due to the mounting location.

c) The safety-related radiation monitors will be provided with uninterruptible power. One set of radiation monitors in the north and south CROAI ducts will be powered from safety related train A, and the other set of radiation monitors will be powered from train B.

d) High Radiation detected by any of the four safety-related monitors will actuate its associated train of control room ventilation system to isolation/recirculation mode.

The above design changes were evaluated for reasonable assurance that the CR ventilation system will be automatically isolated and placed in the recirculation mode of operation as assumed in the AST analyses.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design Basis Accidents

As stated in RG 1.183, Regulatory Position 5.2, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.

The licensee performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183 and SRP Section 15.0.1. The licensee performed AST analyses for the PWR DBAs identified in RG 1.183 that could potentially result in significant CR and offsite doses. These include the loss of coolant accident (LOCA), the fuel-handing accident (FHA), the main steam line break (MSLB) accident, the steam generator (SG) tube rupture accident (SGTR), the reactor coolant pump (RCP) shaft seizure (Locked Rotor) accident (LRA), and the control element assembly (CEA) ejection accident. In addition, the licensee analyzed the inadvertent opening of a main steam safety valve (IOMSSV), which is not addressed in RG 1.183.

The licensee provided the accident specific input assumptions as described in the Numerical Applications, Inc. (NAI) "AST Licensing Technical Report for St. Lucie Unit 1," NAI-110 1-043, Revision 2. These analyses provide for a bounding allowable CR unfiltered air inleakage of 500 cfm. The use of 500 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 Nuclear Energy Institute (NEI) 99-03 and Generic Letter 2003-01.

The DBA radiological source term used in the AST analyses was developed based on a core power level of 2754 megawatts thermal (MWt). The core power level used in the AST analysis of 2754 MWt represents the licensed power of 2700 MWt with a 2% increase to account for measurement uncertainties. The use of 2754 MWt for the AST is acceptable to the NRC staff for use in the full implementation of the AST at St. Lucie Unit 1.

The licensee has performed a full implementation of the AST as defined in RG 1.183. The licensee has determined that the current Technical Information Document (TID)-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactors Sites," accident source term will remain the licensing basis for equipment qualification (EQ).

Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memo dated April 30, 2001 (ADAMS Accession No. ML011210348) and in NUREG-0933, Supplement 25, June 2001 (ADAMS Accession No. ML012190402). The conclusion to Generic Issue 187 "The potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump" states the following: "The NRC staff

concluded that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary." Therefore, in consideration of the cited references, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ at St. Lucie Unit 1.

RG 1.183, Regulatory Position 4.3, Other Dose Consequences, states that: "The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737. . . should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE."

In a letter dated March 18, 2008 (ADAMS Accession No. ML080850561), the licensee provided additional information describing the basis for maintaining the current licensing basis (CLB) radiological dose analyses for post-accident vital area access as described in NUREG-0737, Item II.B.2. The licensee cited the resolution of Generic Issue 187 as described for the justification of maintaining the CLB source term for EQ, as applicable to the radiological dose analyses for post-accident vital area access as well. The licensee asserts, and the NRC staff concurs that since the calculated post-accident vital area access dose rates are not expected to be significantly impacted by the AST during the first 30-days following a LOCA, the conclusions of the shielding study would not change significantly by expressing the mission dose in terms of TEDE. The licensee also stated that since the technical support center (TSC) and the CR share the same habitability envelope, the shielding study consequences for these areas have been addressed in the current AST LAR.

Regarding post-accident sampling capability, as described in NUREG-0737, Item II.B.3, the licensee cited TS amendments No. 174, which eliminated the requirements to have and maintain the post-accident sampling system.

The licensee also cited the resolution of Generic Issue 187 as the basis for maintaining the CLB radiological dose analyses for the accident monitoring instrumentation as described in NUREG-0737, Item II.F.1. The licensee asserts, and the NRC staff confirmed, that the leakage control requirements of NUREG-0737, III.D.1.1 and the CR habitability requirements of NUREG-0737, III.D.3.4 are incorporated into the revised AST radiological analyses.

Regarding emergency response facilities as described in NUREG-0737, III.A.1.2, since the TSC is contained within the CR envelope (CRE) its habitability is evaluated in the current AST LAR. If post-accident conditions warrant, there are plans established for the evacuation and relocation of the operational support. The emergency operations facility is located outside the 10 mile emergency planning zone and therefore, specific post accident dose analyses are not required. Therefore, the NRC staff finds that the licensee has maintained consistency with the NUREG-0737 evaluations while incorporating the AST into the plant licensing basis for DBA dose consequence analyses.

A full implementation of the AST is proposed for St. Lucie Unit 1. Therefore, to support the licensing and plant operation changes discussed in the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183:

1. Loss-of-Coolant Accident
2. Fuel Handling Accident
3. Main Steam Line Break Accident
4. Steam Generator Tube Rupture Accident
5. Reactor Coolant Pump Shaft Seizure Accident
6. Control Element Assembly Ejection Accident
7. Inadvertent Opening of a Main Steam Safety Valve

The DBA dose consequence analyses evaluated the integrated TEDE dose at the exclusion area boundary (EAB) for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low population zone (LPZ) and the integrated dose in the St. Lucie Unit 1 CR were evaluated for the duration of the accident. The dose consequence analyses were performed for the licensee by Numerical Applications, Inc. using the RADTRAD-NAI code. RADTRAD-NAI estimates the radiological doses at offsite locations and in the CR of nuclear power plants as consequences of postulated accidents. The code considers the timing, physical form, and chemical species of the radioactive material released into the environment.

RADTRAD-NAI was developed from the "RADTRAD: Simplified Model for RADionuclide Transport and Removal and Dose Estimation," computer code. NRC sponsored the development of the RADTRAD radiological consequence computer code, as described in NUREG/CR-6604. The RADTRAD code was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff uses the RADTRAD computer code to perform independent confirmatory dose evaluations as needed to ensure a thorough understanding of the licensee's methods. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1 of this SE.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states that "The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 . . . or ORIGEN-ARP . . . ."

In accordance with RG 1.183, Regulatory Position 3.1, the licensee generated the core and worst case fuel assembly radionuclide inventories for use in determining source term inventories using the ORIGEN code version 2.1. The licensee assumed a period of irradiation that was sufficient to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. For the LOCA, in which all 217 of the fuel assemblies are assumed to fail, the licensee based the source term on an average assembly with a core average burnup of 45,000 megawatt days per metric ton of uranium (MWD/MTU) and an average assembly power of 12.691 megawatt thermal (MWt). The licensee based the minimum fuel enrichment on an historical minimum of 3.0 weight percent (w/o) and the maximum fuel enrichment on the TS maximum value of 4.5 w/o. The licensee conservatively assumed that a maximum assembly uranium mass of 424,160 grams applies to all of the fuel assemblies.

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

As stated in Footnote 11 of RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU.

The licensee stated that the St. Lucie Unit 1 core design allows for a maximum of 150 rods, which may exceed these burnup limits. With 217 assemblies in the core, and 176 fuel rods per assembly, the 150 high burnup rods represent 0.393% of the core. The licensee stated that the number of rods exceeding the burnup/linear heat rate for St. Lucie Unit 1 will be limited to 150 rods. For conservatism, the licensee evaluated the AST analyses assuming a total of 1408 rods, the equivalent of 8 fuel assemblies, exceed the high burnup limit. The licensee doubled the activity gap fractions for all rods in 8 assemblies in addition to applying a peaking factor of 1.7 to account for the high burnup rods that exceed the limits specified in RG 1.183. This approach increases the impact of the high burnup fuel rods from the actual 0.393% of the core to 3.687% of the core, which is nearly 10 times larger than the number of affected rods. Doubling the gap release fraction of 3.687% of the core yields a core-wide high burnup adjustment factor of 1.03687. The licensee applied this factor to the release fractions for all events in which fuel damage causes the inventory of the fuel rod gaps to be released into the reactor coolant. For the FHA, in which 100% of the rods in the dropped assembly are assumed to release their gap activity, the licensee addressed the high burnup issue by increasing the gap release fraction of the entire assembly by a factor of two. The NRC staff concludes that the licensee's approach to the evaluation of the high burnup issue at St. Lucie Unit 1 is conservative and therefore acceptable.

The licensee used committed effective dose equivalent (CEDE) and effective dose equivalent (EDE) dose conversion factors (DCFs) from Federal Guidance Reports (FGR) 11 and 12 to determine the TEDE dose as is required for AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance and is therefore acceptable to the NRC staff.

3.1.1 Loss-of-Coolant Accident

The radiological consequence design basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary reactor coolant system (RCS) piping. The accident scenario assumes the deterministic failure of the emergency core cooling system (ECCS) to provide adequate core cooling that results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analyses.

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of the primary RCS piping, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

When using the AST for the evaluation of a design basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.
In the evaluation of the LOCA design basis radiological analysis, the licensee considered dose contributions from the following potential activity release pathways:

- Containment leakage via the secondary containment system.
- Containment leakage bypassing the secondary containment.
- Engineered safety feature (ESF) system leakage into the Auxiliary Building.
- ESF system leakage into the refueling water tank (RWT).
- Containment purge at event initiation.

In addition, the licensee considered potential DBA LOCA dose contributors due to direct shine as part of the CR habitability envelope analysis:

- Contamination of the CR atmosphere by intake and infiltration of radioactive material from the containment leakage and ESF system leakage.

- External radioactive plume shine contribution from the containment and ESF leakage releases with credit for CR structural shielding.

- A direct shine dose contribution from the containment's contained accident activity with credit for both containment and CR structural shielding.

- A direct shine dose contribution from the activity collected on the CR ventilation filters.

3.1.1.1 LOCA Source Term

The licensee followed all aspects of the guidance outlined in RG 1.183, Regulatory Position 3, regarding the core inventory and the release fractions and timing for the evaluation of the LOCA.

The LOCA analysis assumes that iodine will be removed from the containment atmosphere by both containment sprays and natural diffusion to the containment walls. As a result of these removal mechanisms, a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gaseous and soluble fission products, such as iodines and cesium, but not noble gases. The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.

The licensee's application for AST included several supportive analyses, one of them consisting of determination of water pH in the post-LOCA containment sump and RWT. Determination of this pH was a requirement specified in Regulatory Position Item 2 of Appendix A to RG 1.183. The analysis was performed to ensure that particulate iodine generated in the damage core and deposited into the containment sump water during the design-basis accident (DBA LOCA) will not re-evolve beyond the amount recognized for this accident. The licensee based its calculation of pH on the methodology developed in NUREG/CR-5950, "Iodine Evolution and pH Control." This methodology was incorporated in the licensee's procedure for numerical calculation of pH in the plant.

<u>pH in Containment Sump</u>

It was recognized by the licensee that the following chemicals would be released to the containment sump:

- Several chemical species from the damaged core
- Borated water from the primary coolant system (PWR)
- Borated water from the RWT
- Borated water from the safety injection tanks (SIT)
- Sodium hydroxide solution (NaOH)
- Hydrochloric acid (HCl)
- Nitric acid (HNO_3)

Most of these chemicals are acidic and it is expected, therefore, that without addition of basic chemicals the resulting sump water will be lower than a pH of 7. To prevent this occurrence, the licensee added sufficient amount of sodium hydroxide. Since boric acid is a weak acid and sodium hydroxide is a strong base, together they will produce a buffer solution and will keep the

sump water at a relatively steady pH. The licensee was able to demonstrate that, with one exception (a pH of 6.98), the post-LOCA water in the containment sump will remain basic for at least 30 days. This is illustrated below:

- Containment Sump Minimum at 30 Days Post-LOCA pH=8.18
- Containment Sump Maximum at 30 Days Post-LOCA pH=9.88
- Containment Sump Minimum at Recirculation pH=6.98
- Containment Sump Minimum at 1 Hour Post-LOCA pH=7.48

With one exception, all the calculated pH values are above 7. Because of inherent conservatism of the calculation, this exception is acceptable. The licensee's evaluation of containment sump pH determined that the sump pH will be maintained at or above 7. This ensures that particulate iodine deposited into the containment sump water will not re-evolve beyond the amount recognized in the DBA LOCA analysis. Therefore, in accordance with the applicable regulatory guidance, the licensee assumed that the chemical form of the radioiodine released to the containment is 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.

pH in Reactor Water Tank

After a LOCA, most of the borated water from the RWT is released to the containment. However, some of it stays in the RWT. This water mixes with the sump water system is introduced to the RWT through leakage from the engineered safety features (ESF). Since borated water in the RWT has considerably higher concentration of boric acid than sump water, the mixture will be enriched in boric acid and its pH will be lower than in the sump water. This would cause higher conversion of iodine to molecular form and higher release to the environment. The licensee calculated that the RWT pH varied from 4.5 immediately after a LOCA to 5.007 at 30 days post-LOCA.

3.1.1.2 Assumptions on Transport in the Primary Containment

3.1.1.2.1 Containment Mixing, Natural Deposition, and Leak Rate

Section 6.2.1.2 of the St. Lucie Unit 1 updated final safety analysis report (UFSAR) describes the containment vessel as a low leakage steel shell, including all its penetrations, designed to confine the radioactive materials that could be released by accidental loss of integrity of the reactor coolant pressure boundary. Physically, the containment vessel is a right circular cylinder with hemispherical dome and ellipsoidal bottom that houses the reactor pressure vessel, the reactor coolant piping and pumps, the steam generators, the primary coolant pressurizer and pressurizer quench tank, and other branch connections of the reactor coolant system including the safety injection tanks.

In accordance with RG 1.183, the licensee assumed that the activity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the containment. The licensee used the core release fractions and timing, as specified in RG 1.183, with the termination of the release into containment set at the end of the early in-vessel phase.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition. The licensee credited an elemental iodine natural deposition removal coefficient of 2.89 hr^{-1}. The licensee applied the elemental iodine natural deposition removal coefficient of 2.89 hr^{-1} to both the sprayed and unsprayed volume of the containment. The licensee did not credit the removal of organic iodine by natural deposition.

The licensee credited a natural deposition removal coefficient of 0.1 hr^{-1} for all aerosols in the unsprayed region of containment. In addition, the licensee credited a natural deposition removal coefficient of 0.1 hr^{-1} for all aerosols in the sprayed region after spray is terminated at 8 hours.

RG 1.183, Appendix A of Regulatory Position 3.7 states that, "The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate." Accordingly, the licensee assumed a containment leak rate of 0.5% per day for the first 24 hours, after which the containment leak rate is reduced to 0.25% per day for the duration of the accident.

3.1.1.2.2 Containment Spray Assumptions

RG 1.183, Appendix A, Regulatory Position 3.3 states that, "The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown." In addition, SRP Section 6.5.2, III,1, C states, "The containment building atmosphere may be considered a single, well-mixed space if the spray covers regions comprising at least 90% of the containment building space and if a ventilation system is available for adequate mixing of any unsprayed compartments."

For St. Lucie Unit 1, the volume of the sprayed region is 2,155,160 ft^3 and the volume of the unsprayed region is 350,840 ft^3. Since the sprayed region represents approximately 86% of the total containment volume, the licensee used a two volume model to represent the sprayed and unsprayed regions of the containment. The licensee assumed a mixing rate of two turnovers of the unsprayed region per hour. This assumption is in accordance with RG 1.183, Appendix A, Regulatory Position 3.3, which states in part that, "The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified."

In a letter dated September 21, 2004 (ADAMS Accession No. ML042680405), the licensee provided additional information regarding the use of the mixing rate of two turnovers of the unsprayed region per hour. The licensee stated that per the St. Lucie Unit 1 UFSAR, the existing design basis for the mixing rate between the sprayed and unsprayed regions of containment is four unsprayed volumes per hour. There is no reason for this value to change based on adoption of the AST; however, the licensee reduced the value to two per hour to be consistent with RG 1.183. The licensee also noted that NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings," supports significantly higher mixing rates based on the operation of containment sprays.

Using the guidance from SRP 6.5.2, the licensee determined that the aerosol removal rate from the effects of the containment spay system, which actuates 0.017917 hours (64.5 seconds) after the LOCA, is 6.43 per hour until a decontamination factor (DF) of 50 is reached at 2.60 hours post LOCA. After the DF of 50 is reached, the licensee assumed that the aerosol removal rate is reduced by a factor of 10 in accordance with the applicable regulatory guidance.

Using the guidance from SRP 6.5.2, the licensee determined that the elemental iodine removal rate from the effects of the containment spay system, which actuates 0.017917 hours (64.5 seconds) after the LOCA, is in excess of 20 per hour. However, in accordance with the guidance in SRP 6.5.2, the licensee limited the removal rate constant for elemental iodine to 20 per hour. The licensee applied this elemental removal rate in the dose analysis from the time of spray actuation until the maximum allowable DF of 200 is reached at 3.02 hours post LOCA.

3.1.1.3 Assumptions on Dual Containments

Section 6.2.1.2 of the St. Lucie Unit 1 UFSAR describes the shield building, also referred to as the secondary containment, as a medium leakage reinforced concrete structure surrounding the containment vessel. The shield building is designed to provide biological shielding during normal operation and LOCA conditions, environmental protection for the containment vessel from adverse atmospheric conditions and external missiles, and a means for collection and filtration of fission product leakage from the containment vessel following a LOCA. Physically, the shield building is a right circular cylinder with a shallow dome roof.

The licensee assumed that the leakage from primary containment will be collected by the secondary containment and processed by the ESF shield building ventilation system (SBVS) filters prior to release from the plant stack. The licensee credited secondary containment filtration efficiencies of 95% for elemental and organic iodine and 99% for particulates. The licensee assumed that the leakage into the secondary containment is released directly to the environment as a ground-level release prior to the effective drawdown of the secondary containment, which is assumed to be completed at 310 seconds after accident initiation.

The licensee credited the SBVS as being capable of maintaining the shield building annulus at a negative pressure with respect to the outside environment considering the effect of high wind speeds and LOCA heat effects on the annulus as described in UFSAR Section 6.2. The licensee stated that no exfiltration through the concrete wall of the Shield Building is expected to occur.

The licensee did not credit dilution of the primary containment leakage within the secondary containment volume. In addition, the licensee assumed that 9.6% of the primary containment leakage will bypass the secondary containment and be released at ground level without credit for filtration.

3.1.1.4 Assumptions on ESF System Leakage

To evaluate the radiological consequences of ESF leakage, the licensee used the deterministic approach as prescribed in RG 1.183. This approach assumes that except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the containment sump water. Except for iodine, all of the radioactive materials in the containment

sump are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40% of the core inventory of iodine. This amount is the combination of 5% released to the containment sump water during the gap release phase and 35% released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100% of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the containment sump concurrently. ECCS leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections.

For the LOCA analysis of ESF leakage, the licensee used a value of 4510 cubic centimeters per hour (cc/hour), representing two times the current licensing basis value of 2255 cc/hour, as specified in RG 1.183, Appendix A, Regulatory Position 5.2. The licensee assumed that ESF leakage will start at 20 minutes into the event, coinciding with the beginning of the recirculation phase of emergency core cooling, and continue for the 30 day duration of the accident evaluation.

3.1.1.4.1 Assumptions on ESF System Leakage to the Reactor Auxiliary Building (RAB)

RG 1.183, Appendix A, Regulatory Position 5.5, states that, "If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid . . ."

The licensee calculated the fractional iodine release or flashing fraction for ESF leakage as 3.4%. However, the licensee used a flashing fraction of 10%, as prescribed in RG 1.183, for conservatism. The licensee has determined that the pH of the containment sump will not fall below 7.0 for the duration of the accident.

The licensee assumed that the ECCS leakage is released directly into the RAB and released instantaneously into the environment with credit for RAB ECCS area filtration. The licensee credited ECCS area filtration efficiencies of 95% for elemental and organic iodine and 99% for particulates. As noted previously, the licensee assumed that 100% of the particulate activity is retained in the sump water. The licensee did not credit a reduction of activity released to the RAB as a result of dilution or holdup. This is conservative.

In accordance with RG 1.183, for ESF leakage into the RAB, the licensee assumed that the chemical form of the released iodine is 97% elemental and 3% organic.

3.1.1.4.2 Assumptions on ESF System Back leakage to the Refueling Water Tank (RWT)

The licensee evaluated the dose consequence from ECCS backleakage to the RWT by assuming an initial backleakage rate of 2 gpm based upon doubling the current bounding value of 1 gpm. The licensee assumed that this leakage starts at 20 minutes into the event when recirculation begins and continues throughout the 30-day analysis period. Based on sump pH remaining at 7 or above, 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 that 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 licensee determined the time-dependent concentration of the total iodine in the RWT from the tank liquid volume and leak rate. The licensee calculated that the iodine concentration ranged from a minimum value of 0 at the beginning of the event to a maximum value of 4.22E-05 gm-atom per liter at 30 days.

Based upon the backleakage of sump water, the licensee determined that the RWT pH slowly increases from an initial value of 4.5 to a maximum pH of 5.01 at 30 days. Using the time-dependent RWT pH and the total iodine concentration in the RWT liquid space, the licensee determined the amount of iodine that will be converted to the elemental form using the guidance provided in NUREG/CR-5950. The licensee determined that the RWT elemental iodine fraction will range from 0 at the beginning of the event to a maximum of 0.17.

The licensee assumed that the elemental iodine in the liquid region of the RWT will become volatile and partition between the liquid and vapor space in the RWT based upon the partition coefficient for elemental iodine as described in NUREG/CR-5950. The licensee developed a model using the GOTHIC computer code to determine the RWT temperature as a function of time. The licensee used the resulting temperature profile to calculate the elemental iodine partition coefficient as a function of time.

Because the RWT is vented to the atmosphere, there will be no pressure transient in the air region that would affect the partition coefficient. Since no boiling occurs in the RWT, the licensee calculated the flow rate of the released activity from the vapor space within the RWT based upon the displacement of air by the incoming backleakage. The licensee calculated the elemental iodine release rate from the RWT by multiplying the displacement air flow rate times the elemental iodine concentration in the RWT vapor space.

The licensee used the same approach to evaluate the organic iodine release rate from the RWT. The licensee used an organic iodine fraction of 0.0015 from RG 1.183 in combination with a partition coefficient of 1.0 for organic iodine. Consistent with the applicable guidance, the licensee assumed that the particulate portion of the leakage is retained in the liquid phase of the RWT. Therefore, the total iodine release rate is the sum of the elemental and organic iodine release rates.

The NRC staff has reviewed the licensee's analysis of the dose consequence from ECCS backleakage into the RWT and has determined that the analysis follows the applicable regulatory guidance, is conservative, and is therefore acceptable.

3.1.1.5 Assumptions on Containment Purging

The licensee evaluated the radiological effects of containment leakage via open supplemental purge lines, which is assumed to occur for the first 5 seconds of the DBA LOCA. The licensee assumed that 100% of the radionuclide inventory of the RCS is released instantaneously into the containment at the beginning of the event. The containment purge consists of a volumetric flow

rate of 42,000 cfm released to the environment via the plant vent for a period of 5 seconds with no credit for filtration.

During the time period of 5 seconds following accident onset, the licensee assumes that fuel failure has not occurred. This assumption follows the guidance in Table 4 of RG 1.183, which indicates that the initial release of the RCS into containment for a PWR would occur within the first 30 seconds of the accident prior to the onset of fuel damage. Per RG 1.183, the purge release evaluation should assume that 100% of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the LOCA and that this inventory should be based on the TS reactor coolant system equilibrium activity. Accordingly, the licensee based the evaluation of the containment purge contribution based on RCS radionuclide concentrations of 1.0 micro curie per gram (μCi/gm) dose equivalent I-131 (DEI) and 100/E-bar gross activity.

The licensee's current TS definition of DEI references the dose conversion factors for individual isotopes from International Commission on Radiological Protection ICRP 30, which are equivalent to the rounded committed dose equivalent (CDE) thyroid values from FGR-11 for iodine isotopes. With the approval of this LAR, the licensee will change the TS definition of DEI to reference the CDE thyroid values from FGR-11 for iodine isotopes.

3.1.1.6 Control Room Habitability

3.1.1.6.1 CR Ventilation Assumptions for the LOCA

The CR Air Conditioning System (CRACS) and CR Emergency Ventilation System (CREVS) are required to assure CR habitability. RG 1.196 defines the CRE as follows: "The plant area, defined in the facility licensing basis, that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the CR, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident." The design of the CRE and overall descriptions of both the CRACS and the CREVS are contained in Sections 6.4 and 9.4.1 of the St. Lucie Unit 1 UFSAR.

The licensee's assumptions for CR ventilation are in Table 4. During normal plant operation, the CRE is pressurized relative to the surrounding areas at all times with outside air continuously introduced to the CRE at a rate of 750 cfm. For conservatism, the licensee used a value of 920 cfm in the dose analyses.

For the LOCA analysis, the CR ventilation system is initially assumed to be operating in normal mode. The air flow distribution during the normal mode of operation is 920 cfm of unfiltered fresh air with an assumed value of 500 cfm for unfiltered inleakage. After the start of the event, the CR is assumed to be isolated due to a containment isolation actuation signal (CIAS) as a result of a high containment pressure signal. The licensee applied a 50-second delay to account for the time required to reach the CIAS, the time to start the diesel generator and the time for damper actuation. After isolation, the air flow distribution is assumed to consist of 0 cfm of makeup flow from the outside, 500 cfm of assumed 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 into the CR to restore a positive pressure differential and to maintain air quality. Makeup air for CR pressurization is filtered before entering the CR. During this operational mode, the air flow distribution consists of up to 450 cfm of filtered makeup flow, 500 cfm of assumed unfiltered inleakage, and 1550 cfm of filtered recirculation flow.

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

3.1.1.6.2 CR Direct Shine Dose Assumptions

The total CR LOCA dose includes direct shine contributions from the following DBA-LOCA radiation sources:

- Contamination of the CR atmosphere by the intake and infiltration of the radioactive material contained in the radioactive plume released from the facility.
- Direct shine from the external radioactive plume released from the facility with credit for CR structural shielding.
- Direct shine from radioactive material in the containment with credit for both the containment and CR structural shielding.
- Radiation shine from radioactive material in systems and components inside or external to the CRE including radioactive material buildup on the CR ventilation filters.

The licensee asserts, and the NRC staff agrees, that the LOCA shine dose contribution is bounding for all other events. Per Table 6.4-2 of the UFSAR, and all other sources of direct shine dose to the CR can be considered negligible. The 30-day direct shine dose to a person in the CR, considering occupancy, is provided in Table 4 of this SE. For conservatism, the licensee assumed the bounding LOCA CR shine dose for all the DBAs evaluated.

The licensee determined the direct shine dose from three different sources to the CR operator after a postulated LOCA event. These sources are the containment, the CR air filters, and the external cloud that envelops the CR. The licensee used the MicroShield 5 shielding code to determine direct shine exposure to a dose point located in the CR. Each source required a different MicroShield case structure that included different geometries, sources, and materials. A bounding CR filter inventory is established using a case from the sensitivity study with an assumed unfiltered inleakage that produced a CR dose slightly in excess of the 5 rem TEDE dose limit to CR operators without the application of the occupancy factors described in RG 1.183. 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 licensee modeled the external cloud by assigning a source length of 1000 meters in MicroShield to approximate an infinite cloud.

The licensee ran multiple cases 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 CR filter components, as well as the activity inventory in the environment and the containment.

The RADTRAD-NAI sources were then input into the MicroShield case file 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 licensee's CR direct shine dose evaluation are presented in Table 4 of this SE.

The NRC staff finds that the licensee's evaluation of the potential direct shine dose contributions to the CR LOCA dose analysis used conservative assumptions and sound engineering judgment.

3.1.1.7 Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR comply with the reference values and the CR dose criterion provided in 10 CFR 50.67, and the accident specific dose guidelines specified in SRP Section 15.0.1 and RG 1.183. The licensee's assumptions are presented in Tables 4 and 5 and the licensee's calculated dose results are given in Table 1. The NRC staff's review has found that the licensee used analysis, assumptions, and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis LOCA comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.1.2 Fuel-Handling Accident

TID-14844, "Calculations of Distance Factors for Power and Test Reactor Sites" provides the basis for the accident source term used in the currently licensed analysis for St. Lucie Unit 1's FHA, as outlined in Chapter 15.4.3 of the UFSAR. In the CLB for St. Lucie Unit 1, the licensee specifies that all of the fuel rods in a single fuel assembly are damaged when dropped in the event of a FHA. This assumption also applies to the AST analysis provided in the current LAR. A single assembly contains 176 fuel rods. There are a total of 217 assemblies within the St. Lucie Unit 1 reactor core.

For the purpose of implementing AST methodology and supporting the TS changes, as requested by the subject LAR, the licensee reevaluated the FHA using the accident source term pursuant to guidance provided in RG 1.183, Appendix B. This reevaluation of the design basis FHA applied to both the onsite (i.e., CR) and offsite (i.e., EAB and the outer boundary of the LPZ) radiological consequences. The licensee primarily followed the Regulatory Positions noted in RG 1.183 to define the assumptions, parameters, and inputs used in calculating new values for the dose assessment of the FHA.

As noted in the submittal, the licensee considers analysis of the FHA both within the containment and within the fuel handling building (FHB). The dropped fuel assembly inside the containment is assumed to occur with the equipment maintenance hatch fully open and the fuel

assembly drop inside the FHB credits no filtration of the exhaust. The water level above the damaged fuel assembly is maintained at 23 feet minimum for release locations both inside containment (i.e., reactor cavity) and the FHB (i.e., spent fuel pool or SFP). This water cover acts as a barrier to many of the radionuclides released from the dropped assembly. The licensee assumed retention of all non-iodine particulate in the pool, while the iodine releases from the fuel gap into the pool are assumed to be decontaminated by an overall factor of 200. This decontamination factor (DF) results in 0.5% (i.e., 99.5% of the iodine are retained in the pool) of the radioiodine escaping the overlying water with a composition of 70% elemental and 30% organic. In accordance with Regulatory Position 3 of RG 1.183, the licensee assumes 100% of the noble gas exits the pool. All fission products released to the environment occurs over a two 2-hour period. In the subject FHA analysis, the licensee does not credit dilution within the surrounding structures prior to release to the atmosphere. These assumptions follow the guidance of RG 1.183 and are therefore acceptable to the staff.

3.1.2.1 FHA Source Term

For the purpose of this analysis, the licensee assumed a conservative estimate of 72 hours decay time for the movement of fuel, as accounted for in the RADTRAD code analysis. This indicates that any fuel accounted for in the analyzed FHA would have experienced radioactive decay for a period of 72 hours prior to any susceptibility to dropping either in the reactor cavity or SFP. The core fission product inventory that constitutes the source term for this event is the gap activity in the 176 fuel rods assumed to be damaged as a result of the postulated design basis FHA. This is based on a maximum core power level of 2754 MWt, which is 2% greater than the currently licensed thermal power level of 2700 MWt. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident per Appendix B of Regulatory Position 1.2 of RG 1.183.

Guidance provided in RG 1.183 (i.e., Footnote 11) states that the gap activity release fractions, as specified in Table 3 of RG 1.183, have been determined acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU. In order to account for the gap fraction uncertainty in fuel that does not meet the criteria specified in Footnote 11 of RG 1.183 (i.e., high burnup fuel), the licensee conservatively adjusted these gap fractions by a factor of 2.0 as discussed below.

The licensee stated that the St. Lucie Unit 1 core design allows for a maximum of 150 rods, which may exceed the burnup limits specified in Footnote 11 of RG 1.183. Considering 217 assemblies in the core and 176 fuel rods per assembly, the 150 high burnup rods represent 0.393% of the core. The licensee stated that the number of rods exceeding the burnup/linear heat rate for St. Lucie Unit 1 will be limited to 150 rods. Conservatively, the licensee evaluated the AST analyses assuming a total of 1408 rods, the equivalent of eight fuel assemblies, exceed the high burnup limit. The licensee doubled the activity gap fractions for all rods in eight assemblies in addition to applying a peaking factor of 1.70 to account for the high burnup rods that exceed the limits specified in RG 1.183. This approach increases the impact of the high burnup fuel rods from the actual 0.393% of the core to 3.687% of the core, which is nearly 10 times larger than the number of affected rods. Doubling the gap release fraction of

3.687% of the core yields a core-wide high burnup adjustment factor of 1.03687. The licensee applied this factor to the release fractions for all events in which fuel damage causes the inventory of the fuel rod gaps to be released into the reactor coolant. For the FHA, in which 100% of the rods in the dropped fuel assembly are assumed to release their gap activity, the licensee addressed the high burnup issue by increasing the gap release fraction of the entire assembly by a factor of 2.0. The staff concludes that the licensee's approach to the evaluation of the high burnup issue at St. Lucie Unit 1 is conservative and, therefore, acceptable.

3.1.2.2 Transport

Pursuant to guidance provided in RG 1.183, the St. Lucie Unit 1 FHA is analyzed based on the assumption that all of the fission products released from the reactor cavity or SFP are released to the environment over a 2-hour period. The licensee utilized a ground-level release for all scenarios considered for the subject FHA. A drop of a single fuel assembly and a subsequent release from the closest point of the FHB to the CR was found to be the most limiting FHA.

For the FHA occurring inside containment, the licensee assumed that the equipment maintenance hatch is open at the time of the accident and that the release from the containment occurs with no credit taken for containment isolation, no credit for dilution or mixing in the containment atmosphere, and no credit for filtration of the released effluent. For the FHA occurring in the FHB, the licensee also assumed no credit for filtration of the activity released from the SFP water prior to being released to the environment.

As corrected by item 8 of Regulatory Issue Summary (RIS) 2006-04 (ADAMS Accession No. ML053460347), RG 1.183, Appendix B, Regulatory Position 2, should read as follows:

> If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water).

As noted previously, the licensee assumed a minimum water depth of 23 feet covers the underlying damaged fuel assembly in both the reactor cavity and SFP for the FHA analyzed in the subject LAR. The assumed 176 damaged fuel rods in the pool releases 100% of its gap activity within the water, which is scrubbed by the water column as it rises throughout. This scrubbing decontaminates the gap releases with an overall DF of 200. This DF results in 0.5% (i.e., 99.5% of the iodine are retained in the pool) of the radioiodine escaping the overlying water with a composition of 70% elemental and 30% organic iodine. Additionally, 100% of the noble gas is assumed to exit the pool per Regulatory Position 3 of RG 1.183.

3.1.2.3 CR Ventilation Assumptions for the FHA

The licensee's assumptions for CR ventilation are in Table 4. In order to evaluate the CR habitability for the postulated design basis FHA, the licensee assumed three modes of operation for the CR. During normal mode of operation (i.e., prior to CR isolation), there is an even, unfiltered air flow from dual air intakes to the CR at a rate conservatively adjusted to 920 cfm. After the radiation monitors activate the emergency signal, both north and south CR intakes are

closed simultaneously. This occurs approximately 50 seconds into the postulated FHA. Accordingly, the air flow distribution during this post CR isolation mode consists of 0 cfm of outside makeup flow, 500 cfm of assumed unfiltered inleakage, and 2000 cfm of filtered recirculation flow. After 90 minutes from the onset of the accident, the operator acts to open the more favorable CR air intake based on the output of the radiation monitors, maintaining positive pressure and initiating filtered air makeup into the CR. Air flow during this period consists of up to 450 cfm filtered makeup flow, 500 cfm of assumed unfiltered inleakage, and 1550 cfm of filtered recirculation flow. This filtered air makeup continues throughout the remainder of the 30-day (i.e., 720 hours) event. This process is discussed in more detail in Section 3.2.2, "Control Room Atmospheric Dispersion Factors" of this SE. The licensee considered CREVS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 95% for elemental iodine, and 95% for organic iodine.

3.1.2.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated FHA at St. Lucie Unit 1 and concluded that the radiological consequences at the EAB, outer boundary of the LPZ, and CR are within the reference values and the CR dose criterion provided in 10 CFR 50.67 as well as the accident specific dose guidelines specified in SRP 15.0.1. The licensee's assumptions are presented in Tables 4 and 6 and the licensee's calculated dose results are given in Table 1. The NRC staff's review has found that the licensee used analyses, assumptions, and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff finds that all doses estimated by the licensee for the St. Lucie Unit 1 FHA comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are, therefore, acceptable.

3.1.3 Main Steam Line Break Accident

The postulated MSLB accident assumes a double-ended break of a main steam line. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system causes the main steam isolation valves to close and, if the plant is operating at power when the event is initiated, causes the reactor to trip. For the MSLB DBA radiological consequence analysis, a loss of offsite power (LOOP) is assumed to occur shortly after the trip signal. Following a reactor trip and turbine trip, the radioactivity is released to the environment through the SG power operated relief valves (PORVs). Because the LOOP renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

The licensee evaluated the radiological consequences of a MSLB outside containment. In addition, the licensee considered the radiological consequences of a MSLB inside containment. For the MSLB outside containment, the affected SG, hereafter referred to as the faulted SG, rapidly depressurizes and releases the initial contents of the SG to the environment. For the MSLB inside containment, the faulted SG rapidly depressurizes and releases the initial contents of the SG to the containment atmosphere. The MSLB accident is described in Section 15.4.6 of the St. Lucie Unit 1 UFSAR. RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for a PWR MSLB.

The steam release from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The

increased energy removal from the RCS causes a reduction of coolant temperature and pressure. Due to the negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. In addition, the conservative analysis assumes that the most reactive control rod is stuck in its fully withdrawn position after the reactor trip, thereby increasing the possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the safety injection system (SIS).

3.1.3.1 MSLB Source Term

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for a PWR MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TS including the effects of pre-accident and concurrent iodine spiking. The licensee's evaluation indicates that fuel damage is assumed to occur as a result of a MSLB accident. The licensee determined that the activity released from the damaged fuel will exceed that released by the two iodine spike cases. Therefore, the licensee performed the MSLB dose consequence analysis based on the assumption of fuel damage and did not analyze the two iodine spike cases.

The licensee determined the allowable levels of fuel failure for departure from nucleate boiling (DNB) and fuel centerline melt (FCM) 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. In a letter dated March 18, 2008 (ADAMS Accession No. ML080850561), the licensee provided additional information regarding the assumed values of fuel failure used in the AST analyses. The licensee stated that the analyzed fuel failure values used in the AST dose analyses do not represent values that are indicative of those that would be predicted by the core reload analyses. The licensee further stated that typical cycle-specific fuel failures as predicted by core reload analyses are much less than the fuel failure limits established in the AST DBA dose analyses. For instance, the licensee stated that for the MSLB outside containment, the current cycle-specific core reload analysis indicates no fuel damage.

The licensee based the MSLB source term on the total core inventory of the radionuclide groups as described in RG 1.183, Regulatory Position 3.1. The licensee adjusted the source term for the fraction of fuel damaged and applied a radial peaking factor of 1.7 to the inventory of the damaged fuel. 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. The licensee increased the gap release fractions by a factor of 1.03687 to account for high burnup fuel rods as described in Section 3.1 of this SE. For the fraction of the core that is assumed to experience FCM, the licensee applied the guidance provided in RG 1.183, Appendix H, and Regulatory Position 1, to determine the release. This guidance states that the release attributed to fuel melting should be based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and that for the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.

RG 1.183, Appendix E, Regulatory Position 4 states that, "The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide(CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to

iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking." Accordingly, the licensee assumed that the iodine releases to the environment or to the containment from both the faulted SG and the unaffected SG consist of 97% elemental iodine and 3% organic iodine.

Although the release of secondary coolant activity is not specifically addressed in RG 1.183, for the MSLB accident, the licensee evaluated the radiological dose contribution from the release of secondary side activity using the equilibrium secondary side specific activity TS limiting condition for operation (LCO) of 0.1 μCi/gm DEI.

3.1.3.2 Transport

The licensee evaluated two cases for the MSLB; one case is based upon a double-ended break of a main steam line outside of containment, and the second case is based upon a double-ended break of a 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 SG. The postulated MSLB will result in the rapid depressurization of the affected or faulted SG. 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 licensee adjusted the source term for this activity for the fraction of damaged fuel, the non-LOCA fission product gap fractions from Table 3 of RG 1.183 including an adjustment for high burnup fuel, and an adjustment for a radial peaking factor of 1.7. All noble gases associated with this leakage are assumed to be released directly to the environment.

For both cases, the licensee assumed that the primary-to-secondary leak rate is apportioned equally between the SGs at the rate of 0.5 gpm total with 0.25 gpm to any one SG. This is in accordance with proposed change to the accident induced leakage performance criteria of the Steam Generator Program as described in TS Section 6.8.4.1. The licensee has proposed that the criteria be changed from 1.0 gpm total through all SGs and 0.5 gpm through any one SG, to a total of 0.5 gpm through all SGs and 0.25 gpm through any one SG. This proposed change continues to maintain margin to the operational leakage limit specified in the TSs. TS Task Force Traveler (TSTF)-449, Steam Generator Tube Integrity, changed the SG tube leakage TS limit to 150 gallons per day (gpd) per SG, which is roughly equivalent to 0.1 gpm. For the break outside containment, the licensee assumed that the primary-to-secondary leakage into the faulted SG is released directly to the atmosphere. For the break inside containment, the licensee assumed that the faulted SG primary-to-secondary leakage is released into containment. The licensee assumed that all primary-to secondary leakage continues until the faulted SG is completely isolated at 12 hours.

The licensee followed the guidance as described in RG 1.183, Appendix E, and Regulatory Position 5 in all aspects of the transport analysis for the MSLB. RG 1.183, Appendix E, Regulatory Position 5.2, states that, "The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr [pounds per mass per hour]) should be consistent with the basis of the parameter being converted. The [alternate repair criteria (ARC)] leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically

based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3)." The density used by the licensee in converting volumetric leak rates to mass leak rates is based upon RCS conditions, which is consistent with the plant design basis. The licensee used a RCS fluid density to convert the primary-to-secondary leakage from a volumetric flow rate to a mass flow rate, which is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. This methodology follows sound engineering principles and is therefore acceptable to the NRC staff.

RG 1.183, Appendix E, Regulatory Position 5.3, states that, "The primary to secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling (SDC) is in operation and releases from the steam generators have been terminated." In accordance with RG 1.183, the licensee assumed that 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.

In accordance with RG 1.183, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Positions 5.5.1, 5.5.2 and 5.5.3, the licensee assumed that all of the primary-to-secondary leakage into the faulted SG will flash to vapor, and be released to the environment or to the containment with no mitigation. For the unaffected SG that is used for plant cooldown, the licensee assumed that a portion of the leakage would flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following a plant trip when tube uncovery is postulated. The licensee assumed that the primary-to-secondary leakage would mix with the secondary water without flashing during periods of total tube submergence.

The licensee assumed that the postulated leakage that immediately flashes to vapor would rise through the bulk water of the SG into the steam space and be immediately released to the environment or to the containment with no mitigation. For conservatism, the licensee did not credit any reduction for scrubbing within the SG bulk water.

RG 1.183, Appendix E, Regulatory Position 5.5.4, states that, "The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators."

Accordingly, the licensee assumed that the radioactivity in the bulk water of the unaffected SG becomes vapor at a rate that is a function of the steaming rate and the partition coefficient. The licensee used a partition coefficient of 100 for elemental iodine and other particulate radionuclides released from the intact SG.

In accordance with RG 1.183, Appendix E, Regulatory Position 5.6, the licensee evaluated the potential for SG tube bundle uncovery and determined that tube bundle uncovery is postulated to occur in the intact SG for up to 45 minutes following a reactor trip for St. Lucie Unit 1. During this period, the licensee assumed that the fraction of primary-to secondary leakage, which

flashes to vapor, would rise through the bulk water of the SG into the steam space and be immediately released to the environment or the containment with no mitigation. The licensee determined the flashing fraction based on the thermodynamic conditions in the reactor and secondary coolant. The licensee assumed that the leakage that does not flash would mix with the bulk water in the SG.

The licensee determined the steam mass release rates for the intact SG based on a cooldown rate of 100 °F/hr until the RCS temperature reaches 300 °F. The licensee assumed that this cooldown rate is maintained until 8 hours when SDC is assumed to become available. With the availability of SDC, the licensee assumed that the cooldown would continue at a rate of 38 °F/hr until the RCS temperature is reduced to 212 °F.

The licensee assumed that operator action would be taken to restore water level above the top of the tubes in the unaffected SG within 1-hour following a reactor trip. The NRC staff considers that crediting operator action to restore water level above the top of the tubes in the unaffected SG within 1-hour following a reactor trip to be a conservative and acceptable assumption.

The licensee assumed that all secondary releases would occur from the atmospheric dump valve (ADV) with the most limiting atmospheric dispersion factors. For the MSLB inside containment, the licensee assumed that releases from containment through the SBVS are released from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The licensee assumed that 9.6% of the containment leakage is assumed to bypass the SBVS filters and is released unfiltered to the environment as a ground-level release from containment. The licensee assumed an initial leak rate from the containment of 0.5% of the containment air per day. In accordance with applicable guidance, the licensee reduced this leak rate by 50% after 24 hours to 0.25% per day. The licensee credited natural deposition of the radionuclides consistent with the LOCA methodology presented in Section 3.1.1.2.1 of this SE. The licensee did not credit containment sprays for the MSLB analysis.

3.1.3.3 CR Ventilation Assumptions for the MSLB

The licensee's assumptions for CR ventilation are in Table 4. In order to evaluate the CR habitability for the postulated design basis MSLB, the licensee assumed three modes of operation for the control room ventilation system. During the normal mode of operation prior to CR isolation, there is an even, unfiltered air flow from dual air intakes to the CR at a rate conservatively assumed to be 920 cfm with an assumed value of 500 cfm for unfiltered inleakage. After the radiation monitors activate the emergency signal, both the north and south CR intakes are closed simultaneously. This occurs approximately 50 seconds into the postulated MSLB event. Accordingly, the air flow distribution during this post CR isolation mode consists of 0 cfm of outside makeup flow, 500 cfm of assumed unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

After 90 minutes from the onset of the accident, operator action is credited to open the more favorable CR air intake based on the output of the radiation monitors, maintaining positive pressure and initiating filtered air makeup into the CR. Air flow during this period consists of up to 450 cfm filtered makeup flow, an assumed 500 cfm of unfiltered inleakage, and 1550 cfm of filtered recirculation flow. This filtered air makeup continues throughout the remainder of the

30-day accident evaluation period. This process is discussed in more detail in Section 3.2.2, "Control Room Atmospheric Dispersion Factors" of this SE. The licensee assumed CREVS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 95% for elemental iodine, and 95% for organic iodine.

3.1.3.4.1 Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and CR comply with the reference values and the CR dose criterion provided in 10 CFR 50.67 and the accident specific dose guidelines specified in SRP Section 15.0.1 and RG 1.183. The licensee's assumptions are presented in Tables 4 and 7 and the licensee's calculated dose results are given in Table 1. The NRC staff's review has found that the licensee used analysis, assumptions, and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis MSLB comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.1.4 Steam Generator Tube Rupture Accident

The licensee evaluated the radiological consequences of a SGTR accident as a part of the full implementation of an AST. The SGTR event is described in Section 15.4.4 of the St. Lucie Unit 1 UFSAR. The SGTR accident is evaluated based on the assumption of an instantaneous and complete severance of a single SG tube. At normal operating conditions, the leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the charging pumps. For leaks that exceed the capacity of the charging pumps, pressurizer water level and pressurizer pressure decrease and an automatic reactor trip results. The turbine then trips and the main steam dump and bypass valves open, discharging steam directly into the condenser.

The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG. Integrity of the barrier between the RCS and the main steam system is significant from a radiological release standpoint. The radioactivity from the ruptured SG tube mixes with the shell-side water in the affected SG. As stated in the UFSAR, detection of reactor coolant leakage to the steam system is facilitated by radiation monitors in the SG blowdown lines, in the condenser air ejector discharge lines and in the main steam line radiation monitors. These monitors initiate alarms in the CR and alert operators of abnormal activity levels and that corrective action is required.

For the SGTR DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the reactor trip signal. With a LOOP, the cessation 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 atmospheric dump valves (ADVs). The licensee assumed that this direct steam relief continues until the ruptured SG is isolated at 30 minutes. This credited operator action after 30 minutes is a part of the current licensing basis for the SGTR accident.

3.1.4.1 SGTR Source Term

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR accident. If a licensee demonstrates that no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TS. Two radioiodine spiking cases are considered. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For St. Lucie Unit 1, the maximum iodine concentration allowed by TS as a result of an iodine spike is 60 μCi/gm DEI.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as an accident-induced iodine spike or a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For St. Lucie Unit 1, the RCS TS limit for normal operation is 1.0 μCi/gm DEI. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the iodine equilibrium release rate corresponding to the iodine concentration at the TS limit for normal operation. The iodine release rate at equilibrium is equal to the rate at which iodine is lost due to radioactive decay, RCS purification, and RCS leakage. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an SGTR accident. Therefore, consistent with the CLB and regulatory guidance, the licensee performed the SGTR accident analyses for the pre-accident iodine spike case and the concurrent accident iodine spike case. In accordance with regulatory guidance, the licensee assumed that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system. In accordance with regulatory guidance, the licensee assumed that the iodine releases from the SGs to the environment consist of 97% elemental iodine and 3% organic iodine.

Although the release of secondary coolant activity is not addressed in RG 1.183, for the SGTR accident, the licensee evaluated the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 μCi/gm DEI.

3.1.4.2 Transport

The licensee followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5, in all aspects of the transport analysis for the SGTR dose consequence analysis.

The licensee apportioned the primary-to-secondary leak rate is between the SGs as specified by proposed change to TS 6.8.4.1, which is 0.5 gpm total and 0.25 gpm to any one SG. Therefore, the licensee apportioned the SG tube leakage equally between the two SGs.

RG 1.183, Appendix F, Regulatory Position 5.2, states that, "The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications." The density used by the licensee in converting volumetric leak rates to mass leak rates is based upon RCS conditions, which is consistent with the plant design basis. The licensee used a RCS fluid density to convert the primary-to-secondary leakage from a volumetric flow rate to a mass flow rate, which is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. This follows the methodology of RG 1.183 and sound engineering principles and is, therefore, acceptable to the NRC staff.

RG 1.183, Appendix F, Regulatory Position 5.3, states that, "The primary to secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from the unaffected steam generators should be assumed to continue until SDC is in operation and releases from the steam generators have been terminated." The St. Lucie Unit 1 CLB 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 ruptured SG, while primary-to-secondary leakage continues to provide activity for release from the unaffected SG.

The licensee assumed that a portion of the primary-to-secondary ruptured tube flow or break flow through the SGTR will flash to vapor based on the thermodynamic conditions in the RCS and the secondary system. For the unaffected SG used for plant cooldown, the licensee assumed that flashing would occur immediately following the reactor trip when tube uncovery is postulated. The licensee credited operator action to restore water level above the top of the tubes in the unaffected SG within a conservative time of 1-hour following a reactor trip. The licensee assumed that primary-to-secondary leakage would mix with the secondary water without flashing during periods of total tube submergence.

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the ruptured SG by the break flow. A portion of the break flow is assumed to flash to steam because of the higher enthalpy in the RCS relative to the secondary system. The licensee assumed that the flashed portion of the break flow will ascend through bulk water of the SG, enter the steam space of the affected generator, and be immediately available for release to the environment with no credit taken for scrubbing. Although RG 1.183 allows the use of the methodologies described in NUREG-0409 to determine the amount of scrubbing credit applied to the flashed portion of the break flow, the licensee did not credit scrubbing of the activity in the break flow in the ruptured SG.

During the first 0.1053 hours (379.1 seconds) of the event, prior to the reactor trip and the assumed concurrent LOOP, the licensee assumed that all of the SG flow is routed to the condenser. After 379 seconds, the condenser is no longer available due to the assumed LOOP.

The iodine and other non-noble gas isotopes in the non-flashed portion of the break flow are assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable partition coefficient.

In accordance with applicable regulatory guidance of RG 1.183, Appendix E Regulatory Position 5.5.4, the licensee assumed a partition coefficient of 100 for iodine. The licensee assumed that the retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The licensee assumed the same partition coefficient of 100, as used for iodine, for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

In accordance with RG 1.183, Appendix E, Regulatory Position 5.6, the licensee evaluated the potential for SG tube bundle uncovery and determined that tube bundle uncovery is postulated to occur in the intact SG for up to 45 minutes following a reactor trip for St. Lucie Unit 1. During this period, the licensee assumed that the fraction of primary-to secondary leakage that flashes to vapor would rise through the bulk water of the SG into the steam space and be immediately released to the environment or the containment with no mitigation. The licensee determined the flashing fraction based on the thermodynamic conditions in the reactor and secondary coolant. The licensee assumed that the leakage that does not flash would mix with the bulk water in the SG.

3.1.4.3 CR Ventilation Assumptions for the SGTR

The licensee's assumptions for CR ventilation are in Table 4. In order to evaluate the CR habitability for the postulated design basis SGTR, the licensee assumed three modes of operation for the CR ventilation system. During the normal mode of operation prior to CR isolation, there is an even, unfiltered air flow from dual air intakes to the CR at a rate conservatively assumed to be 920 cfm with an additional assumed unfiltered inleakage of 500 cfm. After the radiation monitors activate the emergency signal, both the north and south CR intakes are closed simultaneously. For the SGTR event, the licensee conservatively assumed that the CR isolation signal would be delayed until the release from the ADVs is initiated at approximately 379 seconds. The licensee included an additional 50-second delay to account for the diesel generator start time, fan start, and damper actuation time. Therefore, for the SGTR analysis, the licensee assumed that CR isolation would occur approximately 429 seconds after initiation of the postulated SGTR event. After isolation, the air flow distribution consists of 0 cfm of outside makeup flow, an assumed 500 cfm of unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

After 90 minutes from the onset of the accident, operator action is credited to open the more favorable CR air intake based on the output of the radiation monitors, maintaining positive pressure and initiating filtered air makeup into the CR. Air flow during this period consists of up to 450 cfm filtered makeup flow, an assumed 500 cfm of unfiltered inleakage, and 1550 cfm of filtered recirculation flow. This filtered air makeup continues throughout the remainder of the 30-day accident evaluation period. This process is discussed in more detail in Section 3.2.2, "Control Room Atmospheric Dispersion Factors" of this SE. The licensee assumed CREVS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 95% for elemental iodine, and 95% for organic iodine.

3.1.4.4 Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR comply

with the reference values and CR dose criterion provided in 10 CFR 50.67 and the accident specific dose guidelines specified in SRP Section 15.0.1 and RG 1.183. The licensee's assumptions are presented in Tables 4 and 8 and the licensee's calculated dose results are given in Table 1. The NRC staff's review has found that the licensee used analyses, assumptions, and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis SGTR will comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.1.5 Reactor Coolant Pump Shaft Seizure (Locked Rotor) Accident

Section 15.3.4 of the UFSAR for St. Lucie Unit 1 describes the locked rotor accident (LRA) as an event in which the instantaneous seizure of a single RCP shaft occurs due to mechanical failure. The principal purpose of the RCP is to provide forced coolant flow through the core of the reactor. As a result of the mechanical failure, flow through the affected primary-to-secondary loop is rapidly reduced; ultimately, causing a three-pump system of reactor coolant flow through the core versus a four-pump system. The postulated sequence of events following an LRA is a reactor trip due to the low coolant flow rate, stored heat transferred to the primary coolant, rapid temperature increase in primary RCS, probable fuel damage due to a decrease of initial DNB margin, and SG tube leakage due to a significant pressure differential between the primary and secondary systems. This event is reanalyzed via the AST methodology, as provided in the current LAR. In the submittal, the licensee evaluates the primary-to-secondary release path in the event of a LRA. Considering this release, fission products from the damaged fuel in the St. Lucie Unit 1 reactor core are assumed to mix instantaneously and homogeneously in the primary coolant. Primary coolant activity transfers to the secondary system (i.e., SGs) via SG tube leakage. Primary coolant activity from SG tube leakage together with secondary activity is postulated to be released to the environment via the ADVs.

For the purpose of implementing AST methodology and supporting the TS changes, as requested by the subject LAR, the licensee reevaluated the LRA using the accident source term pursuant to guidance provided in RG 1.183, Appendix G. This reevaluation of the design basis LRA applied to both the CR and offsite (i.e., EAB and outer boundary of LPZ) radiological consequences. The licensee primarily followed the regulatory positions noted in RG 1.183 to define the assumptions, parameters, and inputs used in calculating new values for the dose assessment of the postulated LRA.

3.1.5.1 LRA Source Term

For the purpose of this AST analysis, St. Lucie Unit 1 assumes that 13.7% limits the amount of fuel assemblies that will experience DNB as a result of the LRA (i.e., about 30 damaged fuel assemblies or 5,227 damaged fuel rods). In deriving the source term for the subject LRA, St. Lucie Unit 1 makes assumptions consistent with regulatory positions illustrated in RG 1.183, Appendix G. Per this guidance, St. Lucie Unit 1 assumes that for the release path analyzed in the event of a LRA, all activity released from the breached fuel assemblies mixes both instantaneously and homogeneously throughout the primary coolant system. This activity is assumed to be released to the secondary system via SG tube leakage.

In accordance with RG 1.183, Appendix G, Regulatory Position 4, the licensee assumed that the chemical form of radioiodine released from the breached fuel assemblies consists of 95% CsI, 4.85% elemental iodine, and 0.15% organic iodide. The licensee also assumed that the chemical form of radioiodine released from the SGs to the environmental atmosphere consists of 97% elemental iodine and 3% organic iodide. This speciation is applicable to both the iodine released as a result of fuel damage and the iodine released from the pre-accident equilibrium iodine concentrations in the RCS and in the secondary coolant system.

The core fission product inventory from RG 1.183, Regulatory Position 3.1 constitutes the source term for the reanalyzed LRA. This is based on a maximum core power level of 2754 MWt, which is 2% greater than the currently licensed thermal power level of 2700 MWt with a core average burnup of 45,000 MWD/MTU. The licensee adjusted the core inventory for the fraction of fuel that is assumed to experience clad damage and conservatively applied a radial peaking factor of 1.7.

Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The licensee has determined that the LRA will result in a limited amount of fuel clad damage. Specifically, the St. Lucie Unit 1 LRA analysis assumes that 13.7% of the total of 217 assemblies in the reactor core will experience fuel clad damage as a result of the transient. For the purpose of dose assessment regarding the non-LOCA LRA event, the licensee used the noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel as specified in Table 3 of RG 1.183. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released as a result of the accident per Regulatory Position 1.2 of RG 1.183. Guidance provided in RG 1.183 (i.e., Footnote 11) states that the gap activity release fractions, as specified in Table 3 of RG 1.183, have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU. In order to account for the gap fraction uncertainty in fuel that does not meet the criteria specified in Footnote 11 of RG 1.183 (i.e., high burnup fuel), the licensee conservatively adjusted these gap fractions by a factor of 1.03687 as discussed below.

The licensee stated that the St. Lucie Unit 1 core design allows for a maximum of 150 rods, which may exceed the burnup limits specified in Footnote 11 of RG 1.183. Considering 217 assemblies in the core and 176 fuel rods per assembly, the 150 high burnup rods represent 0.393% of the core. The licensee stated that the number of rods exceeding the burnup/linear heat rate for St. Lucie Unit 1 will be limited to 150 rods. Conservatively, the licensee evaluated the AST analyses assuming a total of 1408 rods, the equivalent of eight fuel assemblies, exceed the high burnup limit. The licensee doubled the activity gap fractions for all rods in eight assemblies in addition to applying a peaking factor of 1.70 to account for the high burnup rods that exceed the limits specified in RG 1.183. This approach increases the impact of the high burnup fuel rods from the actual 0.393% of the core to 3.687% of the core, which is nearly 10 times larger than the number of affected rods. Doubling the gap release fraction of 3.687% of the core yields a core-wide high burnup adjustment factor of 1.03687. The licensee applied this factor to the release fractions for the subject LRA, in which fuel damage causes the inventory of the fuel rod gaps to be released into the reactor coolant.

Additionally, St. Lucie Unit 1 accounts for the TS limited RCS and secondary activity in the calculations. The licensee assumed that the initial RCS activity is at the specified TS limit of 1.0 μCi/gm DEI and 100/E-bar gross activity. Accordingly, the licensee assumes the initial secondary activity is at the TS limit of 0.1 μCi/gm DEI.

3.1.5.2 Transport

Pursuant to guidance provided in RG 1.183, Appendix G, the St. Lucie Unit 1 LRA is analyzed for the primary-to-secondary release path, with subsequent secondary release to the atmosphere via steaming. This analysis is based on the assumption that all of the fission products released from the damaged fuel in the reactor core as a result of the subject LRA are instantaneously and homogeneously mixed throughout the primary coolant. Applicable to these releases, the licensee analyzed the activity subsequently released to the environment via steaming from the ADVs without scrubbing. This released activity consists of the RCS TS equilibrium activity in addition to activity released from the breached fuel. The licensee assumed that the release of noble gases occurs without mitigation or reduction. The licensee used ground-level mode for the secondary release scenario of the LRA.

St. Lucie Unit 1 consists of a two-loop RCS with two SGs. This results in four cold legs (i.e., two per SG) and four RCPs (i.e., one per SG cold leg). The activity released from the primary RCS to the secondary RCS is assumed to occur at a leak rate of 0.25 gpm per SG for a total of 0.50 gpm. This leakage rate was converted from a volumetric flow rate to a mass flow rate using the RCS fluid density based on a RCS cooldown rate of 100 °F per hour until the RCS temperature reaches 300 °F. After 8 hours, SDC is assumed to be available and the cooldown rate is reduced to 38 °F per hour until the RCS reaches a temperature of 212 °F, which occurs at 10.32 hours. Steam releases from the ADVs are assumed to terminate at this point in time (i.e., 10.32 hours). St. Lucie Unit 1 is currently licensed at a SG leakage rate of 0.5 gpm per SG and 1.0 gpm total. Furthermore, the proposed 0.50 gpm total primary-to-secondary leakage rate is assumed to continue until the SG is fully isolated. The time needed to achieve these conditions is assumed to be 12 hours.

If the temperature of the leakage exceeds 212 °F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. The licensee has determined that the tube bundle in the intact SGs may become uncovered for up to 45 minutes following a reactor trip and that less than 5% of the primary-to-secondary leakage will flash to steam while the tube bundle is uncovered. For the LRA analysis, the licensee conservatively assumed that 5% of the primary-to-secondary leakage will flash to steam for a 1-hour period.

Consistent with Regulatory Positions 5.5.1, 5.5.2, and 5.5.3 of RG 1.183, Appendix E, the licensee assumed that all of the primary-to-secondary leakage that does not flash mixes with the bulk water in the SGs. Additionally, in agreement with Regulatory Position 5.5.4, the licensee assumed that the radioactivity in the bulk water of both SGs becomes vapor at a rate that is a function of the steaming rate and the partition coefficient of 100 for iodine and other particulate radionuclides.

3.1.5.3 CR Ventilation Assumptions for the LRA

The licensee's assumptions for CR ventilation are in Table 4. In order to evaluate the CR habitability for the postulated design basis LRA, the licensee assumed three modes of operation for the CR. During normal mode of operation (i.e., prior to CR isolation), there is an even, unfiltered air flow from dual air intakes to the CR at a rate conservatively adjusted to 920 cfm. After the radiation monitors activate the emergency signal, both north and south CR intakes are closed simultaneously. This occurs approximately 50 seconds into the postulated LRA. Accordingly, the air flow distribution during this post CR isolation mode consists of 0 cfm of outside makeup flow, 500 cfm of assumed unfiltered inleakage, and 2000 cfm of filtered recirculation flow. After 90 minutes from the onset of the accident, operator action is credited to open the more favorable CR air intake based on the output of the radiation monitors, maintaining a positive pressure by initiating filtered air makeup into the CR. Air flow during this period consists of up to 450 cfm filtered makeup flow, 500 cfm of assumed unfiltered inleakage, and 1550 cfm of filtered recirculation flow. This filtered air makeup continues throughout the remainder of the 30-day (i.e., 720 hours) accident analysis period. This process is discussed in more detail in Section 3.2.2, "Control Room Atmospheric Dispersion Factors" of this SE. The licensee considered CREVS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 95% for elemental iodine, and 95% for organic iodine.

3.1.5.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated LRA at St. Lucie Unit 1 and concluded that the radiological consequences at the EAB, outer boundary of the LPZ, and CR are within the reference values and CR dose criterion provided in 10 CFR 50.67 and accident specific dose guidelines specified in SRP 15.0.1. The licensee's assumptions found acceptable to the staff are presented in Tables 4 and 9 and the licensee's calculated dose results are given in Table 1. The staff's review has found that the licensee used analysis, assumptions, and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The staff finds that the doses estimated by the licensee for the St. Lucie Unit 1 LRA comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are, therefore, acceptable.

3.1.6 Control Element Assembly Ejection Accident

Section 15.4.5 of the UFSAR for St. Lucie Unit 1 describes the control element assembly (CEA) ejection accident as the mechanical failure of a CEA and drive shaft resulting in a rapid withdrawal of a single CEA from the reactor core. This uncontrolled ejection of a CEA is caused by a sudden circumferential break of either the control element drive mechanism (CEDM) pressure housing or the CEDM nozzle of the reactor vessel head. As a result, the pressure of the RCS acts to fully eject a CEA. The primary consequence of the described mechanical failure is a rapid reactivity insertion together with an adverse core power distribution (i.e., exponential increase in power) leading to a reactor trip and possible fuel rod damage. In the CLB for St. Lucie Unit 1, the licensee considers this event during both hot full power and hot zero power conditions assuming a total CEA ejection time of 0.05 seconds. These cases, with modifications, are reanalyzed for the AST analysis provided in the current LAR. In the submittal, the licensee evaluates two independent release paths in the event of a CEA accident. The first

release path assumes an instantaneous and homogeneous release of fission products from the damaged fuel in the reactor core to the containment atmosphere with successive release to the environment via containment leakage. The second release pathway assumes that all of the activity released from the damaged fuel is fully dispersed in the primary coolant and subsequently released to the secondary system via SG tube leakage. Activity is subsequently released from the secondary side to the environment via steaming from the ADVs.

For the purpose of implementing AST methodology and supporting the TS changes, as requested by the subject LAR, the licensee reevaluated the CEA event using the accident source term pursuant to guidance provided in RG 1.183, Appendix H. This reevaluation of the design basis CEA accident applied to both the CR and offsite (i.e., EAB and outer boundary of the LPZ) radiological consequences. The licensee primarily followed the regulatory positions noted in RG 1.183 to define the assumptions, parameters, and inputs used in calculating new values for the dose assessment of the CEA accident.

3.1.6.1 CEA Ejection Accident Source Term

For the purpose of this AST analysis, St. Lucie Unit 1 assumes in both release scenarios that 9.5% of the fuel rods experience DNB and 0.5% of the fuel will experience FCM as a result of the CEA ejection from the reactor core. In deriving the source term for the subject CEA event, St. Lucie Unit 1 makes assumptions consistent with Regulatory Position 1 of RG 1.183, Appendix H (also found in Regulatory Position 3 of RG 1.183). Per this guidance, St. Lucie Unit 1 assumes the following conditions for the two release paths analyzed in the provided AST analysis:

> For the containment leakage release pathway, it is assumed that in the event of a CEA accident, 100% of the noble gases and 25% of the iodine contained in the assumed fraction of melted fuel are available for release via containment leakage. In addition, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines resides in the fuel gap. All of the activity released as a result of clad damage and core centerline melting is assumed to be released both instantaneously and homogeneously throughout the containment atmosphere.
>
> For the secondary system release pathway, it is assumed that in the event of a CEA, accident, 100% of the noble gases and 50% of the iodine contained in the assumed fraction of melted fuel are released to the RCS. In addition, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines resides in the fuel gap. All of the activity released as a result of clad damage and core centerline melting is assumed to be released both instantaneously and homogeneously throughout the primary coolant system and to be available for release to the secondary system via SG tube leakage.

In accordance with RG 1.183, Appendix H, Regulatory Position 4, the licensee assumed that the chemical form of radioiodine released to the containment atmosphere consists of 95% CsI,

4.85% elemental iodine, and 0.15% organic iodide. The licensee credits effective controls to limit the pH in the containment sump to 7.0 or higher. In agreement with Regulatory Position 5 of RG 1.183, Appendix H, the licensee assumed that the chemical form of radioiodine released from the SGs to the environment consists of 97% elemental iodine and 3% organic iodide.

The core fission product inventory from the LOCA event constitutes the source term for the reanalyzed CEA accident. This is based on a maximum core power level of 2754 MWt, which is 2% greater than the currently licensed thermal power level of 2700 MWt, and a core average burnup of 45,000 MWD/MTU. The licensee adjusted the core inventory for the fraction of fuel that is assumed to experience clad damage and FCM and conservatively applied a radial peaking factor of 1.7.

Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. Following guidance of Regulatory Position 1 of RG 1.183, Appendix H, the licensee assumes that 10% of the core inventory of noble gases and iodine reside in the fuel gap. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released as a result of the accident per Regulatory Position 1.2 of RG 1.183. Guidance provided in RG 1.183 (i.e., Footnote 11) states that the gap activity release fractions, as specified in Table 3 of RG 1.183, have been determined acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU. In order to account for the gap fraction uncertainty in fuel that does not meet the criteria specified in Footnote 11 of RG 1.183 (i.e., high burnup fuel), the licensee conservatively adjusted these gap fractions by a factor of 1.03687 as discussed below.

The licensee stated that the St. Lucie Unit 1 core design allows for a maximum of 150 rods, which may exceed the burnup limits specified in Footnote 11 of RG 1.183. Considering 217 assemblies in the core and 176 fuel rods per assembly, the 150 high burnup rods represent 0.393% of the core. The licensee stated that the number of rods exceeding the burnup/linear heat rate for St. Lucie Unit 1 will be limited to 150 rods. Conservatively, the licensee evaluated the AST analyses assuming a total of 1408 rods, the equivalent of eight fuel assemblies, exceed the high burnup limit. The licensee doubled the activity gap fractions for all rods in eight assemblies in addition to applying a peaking factor of 1.70 to account for the high burnup rods that exceed the limits specified in RG 1.183. This approach increases the impact of the high burnup fuel rods from the actual 0.393% of the core to 3.687% of the core, which is nearly 10 times larger than the number of affected rods. Doubling the gap release fraction of 3.687% of the core yields a core-wide high burnup adjustment factor of 1.03687. The licensee applied this factor to the release fractions for the subject CEA event, in which fuel damage causes the inventory of the fuel rod gaps to be released into the reactor coolant.

Additionally, St. Lucie Unit 1 accounts for the TS limited RCS and secondary system activity in the calculations. The licensee assumed that the initial equilibrium RCS activity is at the specified TS limit of 1.0 μCi/gm DEI and 100/E-bar gross activity. The licensee assumed the initial equilibrium secondary activity is at the TS 3.7.1.4 limit of 0.1 μCi/gm DEI.

3.1.6.2 Transport

Pursuant to guidance provided in RG 1.183, Appendix H, the St. Lucie Unit 1 CEA accident is analyzed for two cases. The first case is based on the assumption that all of the fission products released from the damaged fuel in the reactor core are instantaneously and homogeneously mixed throughout the atmosphere of the containment. The licensee analyzed releases from the containment to the environment that are filtered via the shield building ventilation system (SBVS) and the released activity that bypasses the SBVS. The SBVS is assumed to remove 99% of the particulate activity and 95% of both the elemental and organic iodine activity. The licensee assumed 9.6% of the activity leaked from the containment will bypass the SBVS filters in the CEA accident analysis.

The second case assumes that all of the fission products released from the damaged fuel in the reactor core are completely dissolved in the primary coolant system and are transferred to the secondary system via SG tube leakage. The activity in the secondary system is subsequently released to the environment via the ADVs without credit for SG scrubbing.

The licensee utilized the plant stack as the point of release for the containment scenario crediting SBVS filtration. However, this release was considered ground-level per guidance provided in RG 1.145, discussed in more detail in Section 3.2.3, "Offsite Atmospheric Dispersion Factors" of this SE. A ground-level release mode was also used for the containment releases that bypass the SBVS and for the secondary release scenario.

3.1.6.2.1 Transport from Containment

For containment releases of the CEA accident, the licensee assumed that all activity from the breached fuel would release to and mix instantaneously and homogeneously in the containment volume of 2.506E+06 ft^3. As specified in TS 3.6.1.1 limit, this activity was modeled to leak from the containment to the environment at an initial rate of 0.50 weight percent per day for the first 24 hours, followed by a rate of 0.25 weight percent per day for the remaining 29 days of the 30-day CEA accident analysis period. This assumption is consistent with Regulatory Position 6.2 of RG 1.183, Appendix H.

The licensee credited natural deposition of the released activity inside the containment. This credit was applied to the radionuclides released using a removal coefficient of 0.10 per hour for aerosols and 2.89 per hour for elemental iodine. No credit was applied to the natural deposition of organic iodine or for the removal of activity via containment sprays.

3.1.6.2.2 Transport from the Secondary System

For secondary releases of the CEA accident, the licensee assumed that all activity from the breached and melted fuel would release to and completely mix in the primary coolant system. Subsequently, the released activity is assumed to transfer to the secondary coolant system as a result of SG tube leakage. Releases to the environment occur as a result of steaming via the ADVs. The release of noble gases is assumed to occur without mitigation or reduction. The activity released from the primary-to-secondary system is assumed to occur at a leak rate of 0.25 gpm per SG for a total of 0.50 gpm. This leakage rate was converted from a volumetric flow rate to a mass flow rate using the RCS fluid density based on a RCS cooldown rate of

100 °F per hour until the RCS temperature reaches 300 °F. After 8 hours, SDC is assumed to be available and the cooldown rate is reduced to 38 °F per hour until the RCS reaches a temperature of 212 °F, which occurs at 10.32 hours. Steam releases from the ADVs are assumed to terminate at this point in time (i.e., 10.32 hours).

For both cases, the licensee assumed that the primary-to-secondary leak rate is apportioned equally between the SGs at the rate of 0.5 gpm total with 0.25 gpm to any one SG. This is in accordance with proposed change to the accident induced leakage performance criteria of the Steam Generator Program as described in TS Section 6.8.4.1. The licensee has changed the criteria from 1.0 gpm total through all SGs and 0.5 gpm through any one SG to a total of 0.5 gpm through all SGs and 0.25 gpm through any one SG. This change continues to maintain margin to the operational leakage limit specified in the TSs. TSTF-449, Steam Generator Tube Integrity, changed the SG tube leakage TS limit to 150 gpd per SG, which is roughly equivalent to 0.1 gpm. The proposed 0.50 gpm total primary-to-secondary leakage rate is assumed to continue until the SG is fully isolated. The time needed to achieve these conditions is assumed to be 12 hours.

If the temperature of the leakage exceeds 212 °F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. The licensee has determined that the tube bundle in the intact steam generators may become uncovered for up to 45 minutes following a reactor trip and that less than 5% of the primary-to-secondary leakage will flash to steam while the tube bundle is uncovered. For the CEA analysis, the licensee conservatively assumed that 5% of the primary-to-secondary leakage will flash to steam for a 1 hour period.

Consistent with Regulatory Positions 5.5.1, 5.5.2, and 5.5.3 of RG 1.183, Appendix E, the licensee assumed that all of the primary-to-secondary leakage that does not flash mixes with the bulk water in the SGs. Additionally, in agreement with Regulatory Position 5.5.4 Appendix E of this guidance, it is assumed that the radioactivity in the bulk water of both SGs becomes vapor at a rate that is a function of the steaming rate and the partition coefficient of 100 for iodine and other particulate radionuclides.

3.1.6.3 CR Ventilation Assumptions for the CEA Ejection Accident

The licensee's assumptions on CR ventilation are in Table 4. In order to evaluate the CR habitability for the postulated design basis CEA ejection accident, the licensee assumed three modes of operation for the CR. During normal mode of operation (i.e., prior to CR isolation), there is an even, unfiltered air flow from dual air intakes to the CR at a rate conservatively adjusted to 920 cfm. After the radiation monitors activate the emergency signal, both north and south CR intakes are closed simultaneously. This occurs approximately 50 seconds into the postulated CEA accident. Accordingly, the air flow distribution during this post CR isolation mode consists of 0 cfm of outside makeup flow, 500 cfm of assumed unfiltered inleakage, and 2000 cfm of filtered recirculation flow. After 90 minutes from the onset of the accident, operator action is credited to open the more favorable CR air intake based on the output of the radiation monitors, maintaining a positive pressure by initiating filtered air makeup into the CR. Air flow during this period consists of up to 450 cfm filtered makeup flow, 500 cfm of assumed unfiltered inleakage, and 1550 cfm of filtered recirculation flow. This filtered air makeup continues

throughout the remainder of the 30-day (i.e., 720 hours) accident analysis period. This process is discussed in more detail in Section 3.2.2, "Control Room Atmospheric Dispersion Factors" of this SE. The licensee considered CREVS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 95% for elemental iodine, and 95% for organic iodine.

3.1.6.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated CEA accident at St. Lucie Unit 1 and concluded that the radiological consequences at the EAB, outer boundary of the LPZ, and CR are within the reference values and the CR dose criterion provided in 10 CFR 50.67 and the accident specific dose guidelines specified in SRP 15.0.1. The licensee's assumptions are presented in Tables 4 and 10 and the licensee's calculated dose results are given in Table 1. The staff's review has found that the licensee used analysis, assumptions, and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The staff finds that the doses estimated by the licensee for the St. Lucie Unit 1 CEA ejection accident comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.1.7 Inadvertent Opening of a Main Steam Safety Valve (IOMSSV)

This event is caused by the inadvertent opening of a SG main steam safety valve (MSSV). Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products contained in the primary coolant before the accident are discharged from the primary into the secondary system. The licensee assumed that the SG tubes do not remain covered and, therefore, no credit is taken for scrubbing in the SG. In addition, the licensee did not credit a flashing fraction for the primary leakage into the SGs. As a result, the licensee assumed that all of the leaked RCS radioactivity is released to the outside atmosphere from the secondary coolant system through the SG MSSVs. The licensee assumed that all of the activity initially present in the SGs would be released to the environment over a 2-hour period. The IOMSSV event is described in Section 15.2.11 of the UFSAR.

3.1.7.1 IOMSSV Source term

RG 1.183 does not provide specific guidance for an inadvertent opening of a SG MSSV. Therefore, to analyze this event, the licensee referred to the guidance in RG 1.183, Appendix G, for a PWR LRA that the licensee judged, and the NRC staff agrees, to be closely applicable to the conditions of an IOMSSV.

The licensee has determined that no fuel damage is postulated to occur for this event. Therefore, the source term for this event is the initial RCS and secondary side activity present at the beginning of the event. The licensee assumed the activity from the fuel to be released instantaneously and homogeneously through the primary coolant.

The licensee assumed the initial RCS activity to be at the TS limit of 1.0 μCi/gm DEI and 100/E-bar gross activity. The licensee assumed the initial SG secondary side activity to be at the TS 3.7.1.4 limit of 0.1 μCi/gm DEI. The licensee conservatively assumed that the entire contents of both SGs are assumed to be released to the environment over a 2-hour period.

3.1.7.2 Transport

Following NRC guidance, the licensee assumed that the iodine releases from the SGs to the environment would consist of 97% elemental iodine and 3% organic iodine. The licensee assumed that the primary-to-secondary leak rate is apportioned equally between the SGs as specified by the proposed TS 6.8.4.1 to 0.5 gpm total and 0.25 gpm to any one SG. The licensee determined the density used in converting volumetric leak rates to mass leak rates upon RCS conditions, consistent with the plant design basis.

The licensee assumed that the primary-to-secondary leakage would continue until after SDC has been placed in service and the temperature of the RCS is less than 212 °F. The licensee's analysis assumes a coincident LOOP and that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.

The licensee assumed that the SG tubes do not remain covered throughout this event for St. Lucie Unit 1. Therefore, the iodine and transport model for release from the SGs assumes that both SGs discharge all of their contents. As a result of this assumption, all of the primary-to-secondary leakage is assumed to flash to steam and be released to the environment with no mitigation and all the radioactivity within the bulk water in the SGs is assumed to be released directly to the environment over a 2-hour period.

3.1.7.3 CR Ventilation Assumptions for the IOMSSV

The licensee's assumptions on CR ventilation are in Table 4. In order to evaluate the CR habitability for the postulated design basis IOMSSV, the licensee assumed three modes of operation for the control room ventilation system. During the normal mode of operation prior to CR isolation, there is an even, unfiltered air flow from dual air intakes to the CR at a rate conservatively assumed to be 920 cfm. After the radiation monitors activate the emergency signal, both the north and south CR intakes are closed simultaneously. This occurs approximately 50 seconds into the postulated IOMSSV event. Accordingly, the air flow distribution during this post-CR isolation mode consists of 0 cfm of outside makeup flow, 500 cfm of assumed unfiltered inleakage, and 2000 cfm of filtered recirculation flow.

After 90 minutes from the onset of the accident, operator action is credited to open the more favorable CR air intake based on the output of the radiation monitors, maintaining positive pressure and initiating filtered air makeup into the CR. Air flow during this period consists of up to 450 cfm filtered makeup flow, an assumed 500 cfm of unfiltered inleakage, and 1550 cfm of filtered recirculation flow. This filtered air makeup continues throughout the remainder of the 30-day accident evaluation period. This process is discussed in more detail in Section 3.2.2, "Control Room Atmospheric Dispersion Factors" of this SE. The licensee assumed CREVS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 95% for elemental iodine, and 95% for organic iodine.

3.1.7.4 Conclusion

The licensee evaluated the radiological consequences resulting from the postulated IOMSSV accident and concluded that the radiological consequences at the EAB, LPZ, and CR comply

with the reference values provided in 10 CFR 50.67 and the accident specific dose guidelines specified in SRP Section 15.0.1 and RG 1.183. The NRC staff's review has found that the licensee used analysis, assumptions, and inputs consistent with the most applicable regulatory guidance identified in Section 2.0 of this SE. In the absence of directly applicable guidance, the licensee used conservative assumptions to evaluate this event, which are found to be acceptable to the NRC staff. The licensee's assumptions are presented in Tables 4 and 11 and the licensee's calculated dose results are given in Table 1. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis IOMSSV comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.2 Atmospheric Dispersion Estimates

The licensee generated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of seven limiting DBAs on the CR and offsite EAB and outer boundary of the LPZ exposures at the St. Lucie Plant located 12 miles southeast of Ft. Pierce, FL. The licensee used onsite meteorological data for calendar years 1996 through 2001 as an input to the ARCON96 (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") atmospheric dispersion computer models to calculate St. Lucie Unit 1 χ/Q values for the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, and IOMSSV events. The licensee assumed ground-level releases for all analyzed DBAs and their resulting onsite and offsite atmospheric dispersion factors. The resulting χ/Q values represent a change from those currently presented in Chapter 15 of the St. Lucie Unit 1 UFSAR.

3.2.1 Meteorological Data

The licensee used 6-consecutive years of onsite hourly meteorological data collected during calendar years 1996 through 2001 to generate a file of five annual cycles of data. Particularly, St. Lucie Unit 1 used meteorological data for the last 6-months of 1996, and all of years 1997, 1998, 1999, and 2001. St. Lucie Unit 1 used only the first 6-months of the year 2000 due to the low recovery rate for the last 6-months of 2000's data. These data were applied to generate new ground-level CR and TSC χ/Q values and offsite ground-level χ/Q values for use in the current LAR. The data were provided for staff review in the form of hourly meteorological data files suitable for input into the ARCON96 control room atmospheric dispersion computer code. A joint wind speed, wind direction, and atmospheric stability frequency distribution (joint frequency distribution or JFD) was developed using the 1996 through 2001 data for use in the PAVAN offsite atmospheric dispersion computer code.

The set of meteorological data (1996 through 2001) used in the current LAR atmospheric dispersion analyses was selected based on a review of the data set quality (i.e., completeness and accuracy of the data). Wind speed and wind direction data used in the atmospheric dispersion analyses were measured on the St. Lucie Plant's onsite primary meteorological tower at heights of 10.0 meters and 57.9 meters above ground level (AGL). Temperature sensors provided atmospheric stability data (via temperature difference) as well. The combined data recovery of the wind speed, wind direction, and atmospheric stability data was in the upper 90th percentile during each year of the full data set for measurement levels of 10.0 meters and

57.9 meters. The NRC staff determined there was an overall data recovery rate of 95.3%. The licensee noted that the data collection process was based on the guidance provided by RG 1.23, Rev. 0, "Onsite Meteorological Programs."

The staff performed confirmatory and quality assurance evaluations of the meteorological data presented using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. Assessment of the wind speed and wind direction data showed similar results from year to year. There was an average wind speed of 3.7 meters per second (m/s) at the 10.0 meter height AGL for the meteorological data presented. Similarly, National Oceanic and Atmospheric Administration National Climatic Data Center showed an average (measured during years 1983 through 2001) ground-level wind speed of 3.7 m/s for Vero Beach, FL, which is approximately 12 miles northwest of the St. Lucie, FL area. Winds predominantly blew from the east direction at both the lower and upper measurement level during each of the 5-years.

Wind direction frequency distributions for each measurement channel were reasonably similar from year to year between both measurement heights. Wind speed frequency distributions also showed similarly from year to year for both measurement levels with the highest occurrence of wind data in the 3 to 5 m/s range (~ 44%) at the 10.0 meter level and in the 5 to 10 m/s range (~ 49%) at the 57.9 meter level. These data were generally consistent with that presented in Chapter 2.3 of the St. Lucie Unit 1 UFSAR.

Regarding atmospheric stability, measured as the temperature difference between the 57.9 meter and 10.0 meter levels, the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). This resulted in unstable conditions (A-C stability classes) occurring approximately 28.0%, neutral and slightly stable conditions (D-E stability classes) occurring 64.5%, and stable conditions (F-G stability classes) occurring 7.5% of the time within the 1996 through 2001 period. A comparison of the JFD derived by the NRC staff from the 57.9 and 10.0 meter ARCON96 formatted hourly data with the JFD developed by the licensee for input into the PAVAN atmospheric dispersion model showed good agreement.

For the reasons noted above, the meteorological data presented for years 1996 through 2001 were found acceptable by staff evaluation and are considered adequate for use in making atmospheric dispersion estimates used in the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, and IOMSSV dose assessments performed in support of the current LAR for St. Lucie Unit 1.

3.2.2 Control Room Atmospheric Dispersion Factors

The licensee generated new CR χ/Q values for postulated St. Lucie Unit 1 releases for the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, and IOMSSV events using guidance provided in RG 1.194. These new atmospheric dispersion estimates were calculated using ARCON96. RG 1.194 states that ARCON96 is an acceptable methodology for assessing onsite χ/Q values for use in design basis accident radiological analyses. The staff evaluated the

applicability of the ARCON96 model and determined that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current license amendment request for St. Lucie Unit 1.

The wind speed, wind direction, and atmospheric stability measured at the 10.0 meter and 57.9 meter heights AGL served as input for the CR χ/Q calculations. Other inputs included the release/source height, the CR and TSC receptor heights, and the straight-line distance between the source and intake/receptor, all in meters, the direction between intake to source, in degrees, and the default values of 0.2 meters for surface roughness, 0.5 m/s for minimum wind speed, and sector averaging constant of 4.3 (found in Table A-2 of RG 1.194). No diffuse area sources were used in the estimated χ/Q analysis for the purpose of dose assessment.

Radioactive releases from the seven events were assumed to discharge to the environment via nine different source points: the St. Lucie Unit 1 (1) main stack/plant vent (MS), (2) refueling water tank (RWT), (3) closest point of the fuel handling building, (4) auxiliary building louver (L-7A), (5) auxiliary building louver (L-7B), (6) condenser, (7) closest atmospheric dump valve, (8) closest feedwater line point, and (9) containment maintenance hatch. The release heights for each of these sources are: 56.1 meters, 14.6 meters, 13.2 meters, 11.6 meters, 11.6 meters, 1.6 meters, 16.1 meters, 5.2 meters, and 4.9 meters, respectively. Essentially, all releases were assumed to occur at ground level for the purpose of atmospheric dispersion analyses. The main stack/plant vent (MS) release point was treated as a ground-level release pursuant to RG 1.194. It states that the use of stack release mode is acceptable when the release point is greater than 2.5 times the height of the adjacent structure(s). Thus, the MS is considered an acceptable ground-level release with a height (i.e., 56.1 meters) less than 2.5 times the height of its adjacent reactor building structure (i.e., 62.9 meters AGL). The primary onsite receptors modeled for the St. Lucie Unit 1 atmospheric dispersion evaluations, as noted in Table 2, were the three St. Lucie Unit 1 CR intakes (the north wall CR intake, the south wall CR intake, and the midpoint of the north and south CR intakes) used during three different modes of operation.

The licensee considered three modes of operation for the CR while evaluating all seven DBAs. These seven DBAs also addressed sub-events (e.g., primary leakage and secondary leakage), totaling numerous cases of evaluation. Each of these cases were analyzed for different modes of CR operation: prior to CR isolation (mode 1), during CR isolation (mode 2), and after initiation of filtered air makeup into the CR (mode 3). During normal plant operation (i.e., prior to CR isolation), the CRE is pressurized with an even flow of unfiltered fresh air via the north and south CR intakes of the reactor auxiliary building at a rate of 750 cfm (conservatively adjusted to 920 cfm for the purpose of analysis). The χ/Q values generated from the release point to the closest CR air intake or least favorable CR air intake (i.e., north CR intake) are used during this period. Following an accident, the unfiltered inleakage is assumed to continue until 50 seconds from the onset of the accident. At this point, Beta-scintillation radiation monitors for St. Lucie Unit 1 generate an isolation signal to close both north and south CR intakes simultaneously. The 50 second delay includes 10 seconds for diesel start, 35 seconds for damper actuation, and 5 seconds for instrument response (all times considered conservative measures for the St. Lucie Unit 1 CR design). After both north and south CR intakes close and about 90 minutes into the accident, the CR operators will act to un-isolate the CR and maintain positive pressure by initiating filtered air makeup into the CR. This is done via opening of the CR intake with the least

amount of radiation based upon the output of the radiation monitors. The period between 50 seconds and 90 minutes uses the midpoint CR intake to model onsite atmospheric dispersion. Post-initiation of filtered CR air makeup, which occurs at 90 minutes into the accident, the south CR intake is noted as more favorable in the χ/Q analysis for all accidents. The filtered air makeup is assumed to occur for the duration of the 30-day (i.e., 720 hours) event for the atmospheric dispersion and dose assessments.

The licensee notes that St. Lucie Unit 1 uses CREVS for filtration post onset of a DBA. This system is composed of a filter train with high-efficiency particulate air (HEPA) filters and charcoal adsorbers with two redundant booster centrifugal fans. For the purpose of the DBA assessments, credit for dilution of the releases was only given for the MS pursuant to RG 1.194, Section 3.3.2.3. This guidance allows credit for releases only if the dual air intakes are not in the same 90 degree wind direction window and there are redundant ESF-grade radiation monitors enabled to alarm the CR. At St. Lucie Unit 1, the plant vent releases are not in the same 90 degree window as both the north and south CR intakes and the St. Lucie Unit 1 CR is designed with two radiation monitors in a common skid capable of sampling either outside air intake duct. Prior to CR isolation, this credit allows a ½ reduction to the unfavorable CR intake (i.e., modeled as the north CR intake) χ/Q value considering flow is from the "clean intake" (i.e., modeled as the south CR intake). Post CR isolation and throughout the remainder of the event, it also allows a ¼ dilution credit to the favorable CR intake χ/Q value considering both flow from the "clean intake" and the expectation that the CR operator will make the proper selection of favorable CR intake during an emergency.

The staff qualitatively reviewed the majority of inputs to the ARCON96 calculations and found them consistent with the site configuration drawings and staff practice. Additionally, the staff performed a random confirmatory analysis of the licensee's assessments of control room post-accident dispersion conditions generated using the 1996 through 2001 meteorological data and the ARCON96 model. The staff has concluded that the resulting χ/Q values generated by the licensee are acceptable for use in the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, and IOMSSV onsite dose assessments at St. Lucie Unit 1.

3.2.3 Offsite Atmospheric Dispersion Factors

The licensee used a JFD derived from the 1997 through 2001 wind data measured on the primary meteorological tower at the 10.0 meter elevation height as input to the PAVAN computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," November 1982) to determine EAB and outer LPZ boundary χ/Q calculations for all postulated releases. As noted previously, only the first six months of meteorological data for the year 2000 was used in the analysis. All releases were modeled as ground-level pursuant to guidance provided in RG 1.145, in which no release heights were more than 2.5 times the adjacent structures. Atmospheric stability class was calculated using the temperature difference between the 57.9 meter and 10.0 meter heights on the primary tower.

In the offsite χ/Q determinations, the licensee conservatively assumed a minimum containment cross-sectional area of 1565 m^2 and a containment height of 62.9 meters AGL. The licensee considered an overall site ground-level EAB distance of 1537 meters and outer boundary of LPZ distance of 1585 meters. For the purpose of dose assessment, the 0-2 EAB χ/Q value was

used as an input throughout the entire 30-day (i.e., 720 hours) event for each DBA analyzed to determine the limiting 2-hour EAB dose estimate.

The licensee's offsite χ/Q values, listed in Table 3, represent a change from those used in the current licensing basis. The staff evaluated the inputs and assumptions used in the PAVAN calculations and found these χ/Q values acceptable for use in the analysis of the postulated DBAs and their associated EAB and LPZ dose estimates performed for the current LAR.

3.3 Technical Specification Changes

3.3.1 TS Definitions Section 1.10, "Dose Equivalent I-131"

The licensee has proposed to revise the definition of DEI in section 1.10 to reference 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.

The revision of the definition of DEI to reference FGR-11 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 DEI are used to adjust the initial primary coolant iodine activities for use in the dose calculations. The licensee has chosen to use the CDE thyroid DCFs as opposed to the CEDE DCFs based on the reasoning that the former results in slightly more conservative total iodine concentrations in the primary coolant and, therefore, slightly higher doses.

The intent of the TS on RCS specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. The licensee currently calculates DEI using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole body dose and thyroid dose as done previously. The applicable DCFs for the calculation of the inhalation contribution to TEDE would be the CEDE DCFs. However the numerical difference between using the DCFs for CDE thyroid as opposed to CEDE values for the calculation of DEI is minimal. Therefore, it is acceptable to the NRC staff for the licensee to retain the reference to thyroid dose in the DEI definition and to use the CDE thyroid DCFs from FGR 11. The NRC staff has evaluated the proposed definition of DEI and has determined that the incorporation of either the thyroid CDE or the CEDE DCFs from FGR No.11 in the DEI definition is acceptable.

Proposed revision of Tables 3.3-6 and 4.4-3 to add the CR Isolation Area Radiation Monitors to the applicability of TS 3/4.3.3.1, Radiation Monitoring.

The St. Lucie Unit 1 plant is equipped with automatically actuated and redundant isolation valves provided at each CR outside air intake and exhaust air path so that the CRE is isolated on receipt of an outside air intake high radiation signal. Addition of the CR Isolation Area Radiation Monitors to the applicability of TS 3/4.3.3.1 will provide the requisite controls to support crediting the high radiation automatic isolation function of the radiation monitors in the AST analyses.

The licensee has proposed an alarm setpoint of less than two times background, which is low enough to provide the required sensitivity, but high enough to avoid nuisance alarms. The

proposed action statement ensures that, with less than the minimum number of channels operable, the isolation function is maintained by requiring that the emergency ventilation system be placed in the recirculation mode of operation. The proposed Surveillance Requirements (SRs) are consistent with those for the radiation monitoring instrumentation currently included in the scope of the SR. The NRC staff has reviewed the proposed revision to Tables 3.3-6 and 4.4-3 and found that the revision supports the credit for the high radiation automatic isolation function of the radiation monitors in the AST analyses and is, therefore, acceptable from a dose analysis perspective.

3.3.3 The licensee has proposed to revise SR 4.6.6.1 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 (VFTP) in TS Section 6.8.4.k.

3.3.4 The licensee has proposed to revise SR 4.7.8 to relocate the HEPA filter, charcoal adsorber, and flow rate surveillance test acceptance criteria for the ECCS Area Ventilation System to the VFTP in TS Section 6.8.4.k.

The licensee asserts, and the NRC staff agrees, that the relocation of the HEPA filter, charcoal adsorber, flow rate, and heater surveillance test acceptance criteria for the Shield Building Ventilation System and the ECCS Area Ventilation System, as applicable, to the VFTP provides consistency with the existing format for the CR Emergency Ventilation System filter testing requirements, which are modeled after the format of the VFTP in NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants."

In the revised TS, the filter train operational 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 the applicable TS surveillance limits. The licensee ensures that the testing methodology requirements are met by requiring that the tests be performed in accordance with American Society of Mechanical Engineers (ASME) N510-1989 and American Society for Testing and Materials (ASTM) D3803-1989, as applicable. The licensee ensures that the frequency requirements are met by describing the VFTP as a program that tests at the frequencies specified in RG 1.52, Revision 3.

The licensee has proposed to delete SRs 4.6.6.1.b.4 and 4.7.8.1.b.4 since these requirements for in-place HEPA and charcoal adsorber testing are delineated in the referenced RG 1.52, Revision 3.

The licensee has proposed to delete SRs 4.6.6.1.c. [3].b for the Shield Building Ventilation System and 4.7.8.1.c. [3].b for the ECCS Area Ventilation System. These SRs require dioctylphthalate testing of the HEPA filter banks subsequent to reinstalling the charcoal adsorber tray used for obtaining a carbon sample. There is no equivalent SR in RG 1.52, Revision 3. The licensee asserts, and the NRC staff agrees, that HEPA filters are not affected by carbon sampling or subsequent reinstallation of the adsorber trays. Therefore, the NRC staff agrees that the SRs are not necessary to ensure operability of the HEPA filters.

3.3.5 The licensee has proposed to reduce the leakage rate acceptance criterion for secondary containment bypass leakage paths from the Shield Building Bypass Leakage stated in TS 6.8.4.h, "Containment Leakage Rate Testing Program," from 27% to 9.6%.

The proposed reduction of the acceptance criterion for secondary containment bypass leakage paths via Shield Building Bypass Leakage from 27% to 9.6% increases the safety margin in the plant design and is, therefore, acceptable to the NRC staff. The 9.6% value is supported by plant leakage test results.

3.3.6 The licensee has proposed to revise the HEPA filter and charcoal adsorber test acceptance criteria for the ECCS Area and Shield Building Ventilation. Systems relocated to the VFTP are revised as follows: The filter efficiency test acceptance criteria are increased from 99% to 99.95%; the in-place charcoal adsorber efficiency test acceptance criteria are increased from 99% to 99.95%.

The licensee credits removal efficiency of 99% for particulate matter in the AST accident dose evaluations. Regulatory Position 6.3 of RG 1.52, Revision 3, states that, "To be credited with a 99% removal efficiency for particulate matter in accident dose evaluations, a HEPA filter bank in an ESF atmosphere cleanup system should demonstrate an aerosol leak test result of less than 0.05% of the challenge aerosol at rated flow ±10%."

Regarding in-place testing of the charcoal adsorber, Regulatory Position 6.4 of RG 1.52, Revision 3, states that, "The test should be performed in accordance with Section 11 of ASME N510-1989. The leak test should confirm a combined penetration and leakage (or bypass) of the adsorber section of 0.05% or less of the challenge gas at rated flow ±10%."

The licensee credits removal efficiency of 95% for elemental and organic iodine the AST accident dose evaluations. The licensee ensures through laboratory testing of charcoal samples for all of the credited ESF systems, that the penetration of methyl iodide does not exceed 2.5% when tested in accordance with ASTM D3803-1989. Table 1 of RG 1.52 allows an assigned removal efficiency credit of 95% for elemental and organic iodine when representative charcoal samples are tested in accordance with ASTM D3803-1989 to a penetration limit of less than or equal to 2.5%.

The NRC staff reviewed the revised test acceptance criteria for the HEPA filters and the charcoal adsorbers in the ECCS Area and Shield Building Ventilation Systems to ensure that with the acceptance criteria met the filters and adsorbers will support the credited filtration efficiencies assumed in the AST accident analyses. As described in RG 1.52, Revision 3, the efficiency assumptions allowed are dependent on the test acceptance criteria. The revised acceptance criteria are consistent with the criteria provided in RG 1.52, Revision 3, required to support the assumptions of the AST accident analyses and are therefore acceptable to the NRC staff.

3.3.7 The licensee has proposed to replace the reference to RG 1.52, Revision 2 in the VFTP (TS 6.8.4.k) reference to RG 1.52, Revision 3.

Replacing reference to RG 1.52, Revision 2 with reference to RG 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 AST accident analyses and are therefore acceptable to the NRC staff.

3.3.8 The licensee has proposed to replace the reference to American National Standards Institute (ANSI) N510-1975 with reference to ASME N510-1989 in the SRs for the ECCS Area Ventilation System and the Shield Building Ventilation System (SRs 4.7.8.1 and 4.6.6.1, respectively), as well as in the VFTP (TS 6.8.4.k).

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. Replacing reference to ANSI N510-1975 with reference to ASME N510-1989 is consistent with RG 1.52, Revision 3 and is therefore acceptable to the NRC staff.

3.3.9 The accident induced leakage performance criteria of the SG Program described in TS Section 6.8.4.1 is changed from a total of 1.0 gpm through all SGs and 0.5 gpm through any one SG, to a total of 0.5 gpm through all SGs and 0.25 gpm through any one SG.

The proposed change in the accident induced leakage performance criteria of the licensee's SG Program from a total of 1.0 gpm through all SGs and 0.5 gpm through any one SG, to a total 0.5 gpm through all SGs and 0.25 gpm through any one SG, continues to maintain margin to the operational leakage limit specified in the TS. TSTF-449, Steam Generator Tube Integrity, changed the SG tube leakage TS limit to 150 gpd per SG, which is roughly equivalent to 0.1 gpm. The licensee used a limit of 0.25 gpm per SG in the AST accident analyses, which provides additional margin above the 0.1 gpm TS limit. The limit of 0.5 gpm total leakage through all SGs is consistent with the limit of 0.25 gpm per SG and reflects the maximum total allowable leakage. The proposed change in the accident induced leakage performance criteria is consistent with the assumptions in the AST accident analyses and is therefore acceptable to the NRC staff.

3.4 Radiation Monitoring Instrumentation

The current plant licensing basis permits the use of a single radiation monitor for normal operation. Current control room ventilation system isolation is based on high radiation monitor alarm. With the implementation of AST, credit has been taken for automatic system isolation. The NRC staff raised concerns about the automatic isolation capability in case of a single active failure. To satisfy the staff concerns, the redundant safety-related radiation detection/monitoring instrumentation is being provided for both the north and the south CROAI ducts.

SRP 3.2.2, "System Quality Group Classification," provides the current NRC guidance for quality group classification of systems and components important to safety. SRP 3.2.2 identifies control room ventilation systems as fluid systems important to safety for PWR plants. Per SRP 3.2.2, meeting the relevant requirements of 10 CFR Part 50, Appendix A, GDC 1 and 10, as they relate to structures, systems, and components important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed, is acceptable to satisfy the requirements of SRP 3.2.2.

GDC 1, Quality Standards and Records, states in part: "Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." "A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions."

Title 10 CFR 50.55a, Codes and Standards, states, in part, "Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed."

RG 1.26 states that emergency and normal ventilation systems, "should be designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed."

Radiation monitoring equipment is safety related, seismically mounted and powered by safety related power sources, housed within a structure designed to withstand wind/tornado loadings and missile impacts, and are housed within waterproof structure.

The proposed design changes and the proposed SR will provide adequate assurance that the safety function of the radiation monitors to isolate the control room on high radiation will be satisfactorily performed under all credible postulated failure modes, and the control room doses will be maintained within the limits of 10 CFR 50.67. Therefore, the proposed design and safety classification of the CROAI radiation monitors are adequate to satisfy the applicable intent of SRP 3.2.2, as well as GDC 1 and 10 CFR 50.55a.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (72 FR 49578, dated August 28, 2007). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Attachment: Tables

Date: November 26, 2008

Table 1
St. Lucie Unit 1 Radiological Consequences Expressed as TEDE [1]
(rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR[4]
LOCA	1.1E+00	2.5E+00	4.7E+00
MSLB – Outside containment (1.8% DNB)	3.3E-01	9.0E-01	4.8E+00
MSLB – Outside containment (0.43% FCM)	3.6E-01	9.7E-01	5.0E+00
MSLB – Inside containment (29% DNB)	5.2E-01	1.0E+00	4.9E+00
MSLB – Inside containment (6.1% FCM)	7.6E-01	1.4E+00	4.9E+00
SGTR Pre-accident spike	3.1E-01	3.0E-01	3.0E+00
Dose acceptance criteria	2.5E+01	2.5E+01	5.0E+00
SGTR Concurrent iodine spike	8E-02	8E-02	6E-01
Locked Rotor Accident (13.7% DNB)	2.5E-01	5.4E-01	2.5E+00
IOMSSV	2E-02	2E-02	3E-01
Dose acceptance criteria	2.5E+00	2.5E+00	5.0E+00
FHA - Containment	5.3E-01	5.2E-01	1.2E+00
FHA – Fuel Handling Building	5.3E-01	5.2E-01	3.0E+00
CEA Ejection Containment Release[5]	2.6E-01	5.0E-01	2.7E+00
CEA Ejection Secondary Side Release [5]	2.9E-01	6.3E-01	2.6E+00
Dose acceptance criteria	6.3E+00	6.3E+00	5.0E+00

[1] Total effective dose equivalent
[2] Exclusion area boundary - worst 2-hour dose
[3] Low population zone - Integrated 30 day dose
[4] CR - Integrated 30 day dose - assumed unfiltered inleakage of 500 cfm
[5] Assumes 9.5% DNB and 0.5% FCM

Note: Licensee's dose results are expressed to a limit of two significant figures.

Table 2 (Page 1 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

A. Loss-of-Coolant Accident (LOCA): Containment Leakage - Shield Building Ventilation System (SBVS) and Containment Purge / H_2 Purge

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Stack Vent / North CR Intake*	2.35E-03	---	---	---	---
During CR Isolation	Stack Vent / Midpoint CR Intake*	3.78E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.68E-04	4.55E-04	2.11E-04	1.26E-04	9.25E-05

* Credit for dilution was taken in this case.

B. Loss-of-Coolant Accident (LOCA): Containment Leakage – Shield Building Ventilation System (SBVS) Bypass

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest FW Line Point / North CR Intake	7.30E-03	---	---	---	---
During CR Isolation	Closest FW Line Point / Midpoint CR Intake	3.17E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.75E-03	1.35E-03	5.76E-04	3.94E-04	2.94E-04

Table 2 (Page 2 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

C. Loss-of-Coolant Accident (LOCA): Emergency Core Cooling System (ECCS) Leakage

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Aux. Bldg. Louver L-7B / North CR Intake	4.85E-03	---	---	---	---
During CR Isolation	Aux. Bldg. Louver L-7A / Midpoint CR Intake	5.04E-03	---	---	---	---
After Initiation of Filtered Make-up	Aux. Bldg. Louver L-7A / South CR Intake	3.59E-03	2.94E-03	1.24E-03	8.84E-04	6.91E-04

D. Loss-of-Coolant Accident (LOCA): Refueling Water Tank (RWT) Backleakage

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	RWT / North CR Intake	1.38E-03	---	---	---	---
During CR Isolation	RWT / Midpoint CR Intake	1.33E-03	---	---	---	---
After Initiation of Filtered Make-up	RWT / South CR Intake	1.10E-03	9.30E-04	3.96E-04	2.94E-04	2.28E-04

Table 2 (Page 3 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

E. Fuel Handling Accident (FHA): Containment Release

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Containment Main't Hatch / North CR Intake	1.87E-03	---	---	---	---
During CR Isolation	Containment Main't Hatch / Midpoint CR Intake	1.19E-03	---	---	---	---
After Initiation of Filtered Make-up	Containment Main't Hatch / South CR Intake	8.11E-04	6.11E-04	2.79E-04	1.72E-04	1.28E-04

F. Fuel Handling Accident (FHA): Fuel Handling Building (FHB) Release

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	FHB Closest Wall Point / North CR Intake	4.93E-03	---	---	---	---
During CR Isolation	FHB Closest Wall Point / Midpoint CR Intake	3.26E-03	---	---	---	---
After Initiation of Filtered Make-up	FHB Closest Wall Point / South CR Intake	2.00E-03	1.40E-03	6.36E-04	4.22E-04	3.09E-04

Table 2 (Page 4 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

G. Main Steam Line Break (MSLB): Release from Outside Containment

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest ADV / North CR Intake	6.24E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	2.82E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.61E-03	1.26E-03	5.08E-04	3.60E-04	2.71E-04

H. Main Steam Line Break (MSLB): Release from Inside Containment – Shield Building Ventilation System (SBVS)

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Stack Vent / North CR Intake*	2.35E-03	---	---	---	---
During CR Isolation	Stack Vent / Midpoint CR Intake*	3.78E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.68E-04	4.55E-04	2.11E-04	1.26E-04	9.25E-05

* Credit for dilution was taken in this case.

Table 2 (Page 5 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

I. Main Steam Line Break (MSLB): Release from Inside Containment – Shield Building Ventilation System (SBVS) Bypass

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest FW Line Point / North CR Intake	7.30E-03	---	---	---	---
During CR Isolation	Closest FW Line Point / Midpoint CR Intake	3.17E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.75E-03	1.35E-03	5.76E-04	3.94E-04	2.94E-04

J. Steam Generator Tube Rupture (SGTR)

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	**Prior to Turbine Trip** Condenser/ North CR Intake	2.47E-03	---	---	---	---
	After Turbine Trip Closest ADV / North CR Intake	6.24E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	2.82E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.61E-03	1.26E-03	5.08E-04	3.60E-04	2.71E-04

Table 2 (Page 6 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

K. Locked Rotor

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest ADV / North CR Intake	6.24E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	2.82E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.61E-03	1.26E-03	5.08E-04	3.60E-04	2.71E-04

L. Control Element Assembly (CEA) Ejection: Secondary Release

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest ADV / North CR Intake	6.24E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	2.82E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.61E-03	1.26E-03	5.08E-04	3.60E-04	2.71E-04

Table 2 (Page 7 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

M. Control Element Assembly (CEA) Ejection: Inside Containment Leakage - Shield Building Ventilation System (SBVS)

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Stack Vent / North CR Intake*	2.35E-03	---	---	---	---
During CR Isolation	Stack Vent / Midpoint CR Intake*	3.78E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.68E-04	4.55E-04	2.11E-04	1.26E-04	9.25E-05

* Credit for dilution was taken in this case.

N. Control Element Assembly (CEA) Ejection: Inside Containment Leakage - Shield Building Ventilation System (SBVS) Bypass

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest FW Line Point / North CR Intake	7.30E-03	---	---	---	---
During CR Isolation	Closest FW Line Point / Midpoint CR Intake	3.17E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.75E-03	1.35E-03	5.76E-04	3.94E-04	2.94E-04

Table 2 (Page 8 of 8)

St. Lucie Unit 1
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

O. Inadvertent Opening of a Main Steam Safety Valve (IOMSSV)

Operation Mode	Release/ Receptor Pair	χ/Q Values (sec/m^3)				
		0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Prior to CR Isolation	Closest ADV / North CR Intake	6.24E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	2.82E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.61E-03	1.26E-03	5.08E-04	3.60E-04	2.71E-04

Table 3

St. Lucie Unit 1
Offsite Atmospheric Dispersion Factors (χ/Q Values)

Offsite Dose Location		χ/Q Values* (sec/m^3)				
		0 to 2 Hours	0 to 8 Hours	8 to 24 Hours	24 to 96 Hours	96 to 720 Hours
Ground Release	**EAB**	1.03E-04	---	---	---	---
	LPZ	9.97E-05	5.47E-05	4.05E-05	2.11E-05	8.29E-06

*Note that *all* releases are assumed to be ground-level pursuant to RG. 1.145. The 0-2 hour EAB χ/Q value was used throughout the entire design-basis accident (DBA).

Note: Licensee's χ/Q results are expressed to a limit of three significant figures.

Table 4
St. Lucie Unit 1 Control Room Data and Assumptions and Direct Shine Results

Control Room Volume	62,318 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	920 cfm
Assumed unfiltered Inleakage	500 cfm
Emergency Operation	
Isolation Mode (50 seconds isolation time until 90 minutes):	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	2000 cfm
Unfiltered Make-up Flow Rate	0 cfm
Assumed unfiltered Inleakage	500 cfm
Filtered Make-up Mode (90 minutes until 30 days):	
Filtered Make-up Flow Rate	450 cfm
Filtered Recirculation Flow Rate	1550 cfm
Unfiltered Make-up Flow Rate	0 cfm
Assumed unfiltered Inleakage	500 cfm
Filter Efficiencies	
Particulates	99%
Elemental iodine	95%
Organic iodine	95%
CR operator breathing rate	
0 - 720 hours	3.5E-04 m^3/sec
CR occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4
LOCA CR Direct Shine Dose	
Containment	0.03 rem
Filters	0.18 rem
External Cloud	0.07 rem
Total	**0.28 rem**

Table 5 (Page 1 of 3)
St. Lucie Unit 1 Data and Assumptions for the LOCA

Core Power level	2754 MWt (2700 +2%)	
Core Average Fuel Burnup	45,000 MWD/MTU	
Fuel Enrichment	3.0 - 4.5 weight percent (w/o)	
Initial RCS Equilibrium Activity in coolant blowdown	1.0 microCi/gm DEI and 100/E-bar	
Volumetric flow rate due to open purge valves	42,000 cfm	
Duration of flow through open purge valves	5 seconds	
Primary containment leak rate		
0 - 24 hours	0.5% (by weight)/day	
24 - 720 hours	0.25% (by weight)/day	
Elemental iodine wall deposition coefficient (0-720 hours)	2.89 hr^{-1}	
Particulate natural deposition removal coefficient	Unsprayed region	Sprayed region
0 - 8 hours	0.1 hr^{-1}	0 hr^{-1}
8 - 720 hours	0.1 hr^{-1}	0.1 hr^{-1}
Primary containment volume sprayed region	2,155,130 ft^3	
Primary containment volume unsprayed region	350,840 ft^3	
Flow rate between sprayed and unsprayed regions	11,695 cfm	
Elemental spray removal coefficients		
0.017917 - 3.02 hours	20 hr^{-1}	
3.02 - 720 hours	0 hr^{-1}	
Particulate spray removal coefficients		
0.017917 - 2.60 hours	6.43 hr^{-1}	
2.60 - 8 hours	0.64 hr^{-1}	
8 - 720 hours	0 hr^{-1}	
Volume of water in containment sump (minimum)	55,460 ft^3	
ECCS Leakage to RAB (2 times allowed limit)	4510 cc/hr	
ECCS Flashing fraction		
Calculated	7.5%	
Used for dose determination	10%	
Chemical form of released iodine from ECCS leakage		
Particulate	0%	
Elemental	97%	
Organic	3%	
ECCS area filter efficiencies		
Elemental	95%	
Organic	95%	
Particulate	99%	

Table 5 (Page 2 of 3)
St. Lucie Unit 1 Data and Assumptions for the LOCA

Initial RWT liquid inventory	38,842 gallons
ECCS leakage into RWT (2 times allowed value)	2 gpm
Flashing fraction for leakage into RWT	0 %
Time dependent RWT pH values	
Selected times in hours	RWT pH
0.00	4.500
9.72	4.512
22.22	4.528
97.22	4.614
720.00	5.007
Time dependent RWT total iodine concentration (gm-atom/liter)	
Selected times in hours	RWT Iodine concentration
0.00	0.00E+00
9.72	1.73E-06
22.22	3.88E-06
97.22	1.41E-05
720.00	4.22E-05
Time dependent RWT liquid temperature	
Selected times in hours	Temperature (°F)
0.00	100.0
9.72	100.0
22.22	100.6
97.22	105.0
720.00	105.6
Time dependent RWT elemental iodine fraction	
Selected times in hours	Elemental iodine fraction
0.00	0.00E+00
9.72	3.47E-02
22.22	6.88E-02
97.22	1.54E-01
720.00	1.13E-01
Time dependent RWT partition coefficient (PC)	
Selected times in hours	Elemental iodine PC
0.00	45.65
9.72	45.65
22.22	45.13
97.22	41.50
720.00	41.03

Table 5 (Page 3 of 3)
St. Lucie Unit 1 Data and Assumptions for the LOCA

LOCA Adjusted iodine release rate from RWT

Time (hours)	Iodine release rate (cfm)
0.33	4.800E-07
4.17	3.700E-06
11.11	1.444E-05
22.22	1.300E-04
111.11	4.437E-04
305.56	6.429E-04
402.78	6.612E-04
500.00	6.568E-04
597.22	6.404E-04
694.44	5.615E-04

Secondary containment filter efficiency	
Particulate	99%
Elemental iodine	95%
Organic iodine	95%
Secondary containment drawdown time	310 seconds
Secondary containment bypass fraction	9.6%
Containment purge filtration efficiency	0%

Transport assumptions

Secondary containment prior to drawdown	Nearest containment penetration to CR
Secondary containment after drawdown	Plant stack
Secondary containment bypass leakage	Nearest containment penetration to CR
ECCS leakage	ECCS exhaust louver
RWT backleakage	RWT
Containment purge	Plant stack

Table 6
St. Lucie Unit 1 Data and Assumptions for the FHA

Core thermal power level	2754 MWt
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
Minimum post shutdown fuel handling time (decay time)	72 hours
High burnup fuel adjustment factor	2.0
Minimum pool water depth	23 feet
Fuel clad damage gap release fractions (2 times RG 1.183, Table 3)	
I-131	16%
Remainder of halogens	10%
Kr-85	20%
Remainder of noble gases	10%
Alkali metals	24% (remains in pool water)
Pool DF	
Noble gases and organic iodine	1
Aerosols	Infinite
Elemental iodine (23 ft of water cover)	285
Overall iodine (23 ft of water cover)	200 (effective DF)
Chemical form of iodine in pool	
Elemental	99.85%
Organic	0.15%
Chemical form of iodine above pool surface	
Elemental	70%
Organic	30%
Duration of release to the environment	2 hour release
Control room ventilation assumptions	See Table 4

Table 7 (Page 1 of 2)
St. Lucie Unit 1 Data and Assumptions for the MSLB Accident

Core Power level	2754 MWt (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 - 4.5 weight percent (w/o)
Maximum radial peaking factor	1.7
Percent DNB for MSLB outside containment	1.8%
Percent DNB for MSLB inside containment	29%
Percent FCM for MSLB outside containment	0.43%
Percent FCM for MSLB inside containment	6.1%
Initial RCS Equilibrium Activity	1.0 microCi/gm DEI and 100/E-bar
Secondary coolant iodine activity	0.1 microCi/gm DEI
High burnup fuel adjustment factor	1.03687
Primary to secondary leak rates	0.25 gpm per SG
RCS density based on RCS conditions	5.9 – 7.84 lbm/gallon conditions
Time to terminate SG tube leakage	12 hours
Reactor coolant system (RCS) mass	411,500 lbm (minimum)
SG secondary side mass assumptions	
Intact SG	105,000 lbm (minimum)
Faulted SG	205,000 lbm (maximum)
Time to reach 212 ˚F terminating steam release	10.32 hours
Intact SG steam release rate in lbm/min for time interval in hours	
0 - 0.25	7900
0.25 – 0.50	4196
0.50 – 0.75	4707
0.75 – 1.0	5362
1.0 – 1.5	5028
1.5 – 2.25	4725
2.25 – 4.0	3924
4.0 – 8.0	2558
8.0 – 10.32	3094
10.32 – 720	0
SG secondary side iodine partition coefficients	
Intact SG	100
Faulted SG	1(none)
Chemical form of iodine released from the secondary side	
Particulate	0%
Elemental	97%
Organic	3%

Table 7 (Page 2 of 2)
St. Lucie Unit 1 Data and Assumptions for the MSLB Accident

Credit for scrubbing within the SG bulk water	None
Intact SG tube uncovery following reactor trip	
Time until tube recovery	1 hour
Flashing fraction	5 %
Containment volume	2.506E+06 ft^3
Containment leakage rate	
0 to 24 hours	0.5% (by weight)/day
24 – 720 hours	0.25% (by weight)/day
Credit for containment sprays	None
Containment natural deposition coefficients	
Aerosols	0.1 hr^{-1}
Elemental iodine	2.89 hr^{-1}
Organic iodine	None
Secondary containment filter efficiency	
Particulate	99%
Elemental iodine	95%
Organic iodine	95%
Secondary containment drawdown time	310 seconds
Secondary containment bypass fraction	9.6%
Control room ventilation assumptions	See Table 4

Mass rate of Steam Generator Tube Leakage for all applicable DBAs (lbm/min)

Time (hours)	Tube leakage per SG	Total SG tube leakage
0.00 – 0.50	1.47	2.94
0.50 – 1.00	1.52	3.05
1.00 – 1.50	1.62	3.25
1.50 – 2.00	1.71	3.42
2.00 – 2.50	1.78	3.57
2.50 – 3.00	1.85	3.70
3.00 – 3.50	1.90	3.80
3.50 – 9.69	1.92	3.83
9.69 – 12.0	1.96	3.91
12.0 – 720	0	0

Table 8 (Page 1 of 2)
St. Lucie Unit 1 Data and Assumptions for the SGTR Accident

Core power level	2700 MWt (2700 + 2%)
Initial RCS equilibrium activity	1.0 µCi/gm DEI, 100/E-bar
Initial secondary side equilibrium activity	0.1 µCi/gm DEI
Initial maximum RCS equilibrium activity	1.0 µCi/gm DEI
Maximum pre-accident spike iodine concentration	60 µCi/gm DEI
Accident initiated iodine spike appearance rate	335 times equilibrium rate
Duration of accident initiated spike	8 hours
Break flow flashing fraction	
Prior to reactor trip	17%
Following reactor trip	5%
Time to terminate break flow	30 minutes
Primary to secondary SG tube leakage rate	0.25 gpm per SG
RCS density based on RCS conditions	5.9 – 7.8 lbm/gallon
Time to terminate SG tube leakage	12 hours
Time to recover intact SG tubes	1 hour
SG secondary side iodine 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	411,500 lbm
Concurrent iodine spike	438,843 lbm
Secondary coolant system mass	
Minimum for SG tube leakage	105,000 lbm
Maximum for secondary side release	205,000 lbm

SGTR integrated mass releases in lbm during time period in hours used for Dose Analysis

			Steam Release to Atmosphere	
Event @ Initial Time	Time (Hours)	Break flow	Ruptured SG	Intact SG
SGTR	0 to 0.1053	104,660	661,842	656,568
Rx Trip LOOP	0.1053 – 0.5		88,352	86,821
Break flow terminated	0.5 – 2.0	0	0	601,096
RSG PORV BV closed	2.0 – 8.0	N/A	N/A	876,233
Flashing in RSG ends	8.0 – 10.32	N/A	N/A	31.09 lbm/min

Table 8 (Page 2 of 2)
St. Lucie Unit 1 Data and Assumptions for the SGTR Accident

RCS Iodine Inventory (Ci) for 8-hr concurrent spike with an appearance rate factor of 335

Isotope	Appearance rate (Ci/min)	8 hour total (Ci)
I-131	160.4	77,013
I-132	116.0	55,700
I-133	266.0	127,670
I-134	133.3	63,967
I-135	169.8	81,488

RCS Iodine concentrations for SGTR pre-existing spike of 60 µCi/gm DEI

I-131	47.5
I-132	13.1
I-133	67.8
I-134	7.42
I-135	32.3

SG secondary side iodine partition coefficients

Flashed tube flow	None
Non-flashed tube flow	100

Chemical form of iodine released from SGs

Particulate	0%
Elemental	97%
Organic	3%

Control room ventilation assumptions	See Also Table 4
Isolation time (total)	429.2 seconds
Start of release from ADVs	379.2 seconds
Delay for DG start, fan start and dampers	50 seconds

Table 9
St. Lucie Unit 1 Data and Assumptions for the Locked Rotor Accident

Core Power level	2754 MWt (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 - 4.5 weight percent (w/o)
Maximum radial peaking factor	1.7
Percent of fuel rods in DNB	13.7%
High burnup fuel adjustment factor	1.03687
Initial RCS equilibrium activity	1.0 µCi/gm DEI, 100/E-bar
Initial secondary side equilibrium activity	0.1 µCi/gm DEI
RCS density based on RCS conditions	5.9 – 7.8 lbm/gallon
Total primary to secondary leak rate	0.5 gpm
Time to terminate SG tube leakage	12 hours
Time to recover SG tubes following Rx trip	1 hour
Flashing fraction	5%
Time to reach 212 ˚F terminating steam release	10.32 hours
RCS mass – minimum used to maximize dose	411,500 lbm
SG secondary side mass	
Minimum for SG leakage	105,000 lbm/SG
Maximum for secondary side release	205,000 lbm/SG
SG secondary side iodine partition coefficients	
Flashed tube flow	1(none)
Non-flashed tube flow	100

Locked rotor accident steam release rates (lbm/min) for time period (hrs)

Event	Time (Hours)	SG release rate (lbm/min)
LRA	0.00 – 0.25	7900
	0.25 – 0.50	4196
	0.50 – 0.75	4707
	0.75 – 1.00	5362
	1.00 – 1.50	5028
	1.50 – 2.25	4725
	2.25 – 4.00	3924
	4.00 – 8.00	2558
	8.00 – 10.32	3094

Control room ventilation assumptions	See Table 4

Table 10 (Page 1 of 2)
St. Lucie Unit 1 Data and Assumptions for the CEA Ejection Accident

Core Power level	2754 MWt (2700 + 2%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 - 4.5 weight percent (w/o)
Maximum radial peaking factor	1.7
Percent of fuel rods in DNB	9.5%
Percent of fuel rods with FCM	0.5%
Initial RCS equilibrium activity	1.0 μCi/gm DEI, 100/E-bar
Initial secondary side equilibrium activity	0.1 μCi/gm DEI
High burnup fuel adjustment factor	1.03687
Total primary to secondary leak rate	0.5 gpm
RCS density based on RCS conditions	5.9 – 7.8 lbm/gallon
Time to terminate SG tube leakage	12 hours
Time to recover SG tubes following reactor trip	1 hour
Flashing fraction	5%
SG secondary side iodine partition coefficients	
Flashed tube flow	1(none)
Non-flashed tube flow	100
Time to reach 212 ˚F terminating steam release	10.32 hours
RCS mass – minimum used to maximize dose	411,500 lbm
SG secondary side mass	
Minimum for SG leakage	105,000 lbm/SG
Maximum for secondary side release	205,000 lbm/SG
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%
Control room ventilation assumptions	See Table 4

Table 10 (Page 2 of 2)
St. Lucie Unit 1 Data and Assumptions for the CEA Ejection Accident

Containment volume	2.506E+06 ft^3
Containment leakage rate	
0 to 24 hours	0.5% (by weight)/day
24 – 720 hours	0.25% (by weight)/day
Secondary containment filter efficiency	
Particulate	99%
Elemental iodine	95%
Organic iodine	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
Credit for containment sprays	None

CEA ejection accident steam release rates (lbm/min) for time period (hrs)

Time	SG release rate
0.00 – 0.25	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.00	5600
1.00 – 1.50	5250
1.50 – 2.25	4934
2.25 – 4.00	4098
4.00 – 8.00	2671
8.00 – 10.32	3247

Table 11
St. Lucie Unit 1 Data and Assumptions for the IOMSSV

Core Power level	2754 MWt (2700 + 2%)
Initial RCS equilibrium activity	1.0 μCi/gm DEI, 100/E-bar
Initial secondary side equilibrium activity	0.1 μCi/gm DEI
Total primary to secondary leak rate	0.5 gpm
RCS density based on RCS conditions	5.9 – 7.8 lbm/gallon
Time to terminate SG tube leakage	12 hours
Secondary side mass release to environment	Entire inventory in 2 hours
SG secondary side iodine partition coefficients	1(none)
Maximum secondary side mass	205,000 lbm/SG
Control room ventilation assumptions	See Table 4