September 29, 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 2 - ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM (TAC NO. MD6202)

Dear Mr. Stall:

The Commission has issued the enclosed Amendment No. 152 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists 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, Letdown Line Break, and Feedwater Line Break.

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,

/RA/

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

Docket No. 50-389

Enclosures:
1. Amendment No. 152 to NPF-16
2. Safety Evaluation

cc w/enclosures: See next page

September 29, 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 2 - ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM (TAC NO. MD6202)

Dear Mr. Stall:

The Commission has issued the enclosed Amendment No.152 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists 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, Letdown Line Break, and Feedwater Line Break.

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

Docket No. 50-389

Enclosures:
1. Amendment No. 152 to NPF-16
2. Safety Evaluation

cc w/enclosures: See next page

Distribution:
PUBLIC
LPL2-2 R/F
RidsNrrPMBMozafari
RidsOgcRp
JParillo
RidsNrrDraAadb
RidsNrrDeEmcb
RidsNrrDorlLpl2-2
RidsNrrLABClayton
RidsNrrDirsItsb
RidsNrrDorlDpr
NGreene
RidsNrrDeEicb
RidsAcrsAcnw&mMailCenter
G. Hill, OIS (2 Hard Copies)
RidsRgn2MailCenter
BLee
RidsNrrDssScvb
RidsNrrDciCsgb
KParczewski
CBasavarju

ADAMS Accession Number: ML082060400 NRR-058 *By memo
Pkg: ML082060465 TS: ML082410327

OFFICE	LPL2-2/PM	LPL2-2/LA	DRA/AADB/BC		EICB/BC*
NAME	BMozafari	BClayton	RTaylor		WKemper
DATE	09/26/08	09/25/08	5/16/08		6/30/08
OFFICE	EMCB/BC*	CSGB/BC*	SCVB/BC*	OGC	LPL2-2/BC
NAME	KManoly	AHiser	RDenning	KSexton	TBoyce
DATE	6/16/08	6/5/08	4/15/08	09/26/08	09/29/08

OFFICIAL RECORD COPY

cc:
Mr. Mano Nazar
Senior Vice President
and Nuclear Chief Operating Officer
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408

Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
P.O. Box 6090
Jensen Beach, Florida 34957

Craig Fugate, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

M. S. Ross, Managing Attorney
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Marjan Mashhadi, Senior Attorney
Florida Power & Light Company
801 Pennsylvania Avenue, NW.
Suite 220
Washington, DC 20004

Mr. Douglas Anderson
County Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

Mr. Gordon L. Johnston
Site Vice President
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957-2000

Mr. Christopher R. Costanzo
Plant General Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. Eric Katzman
Licensing Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Don E. Grissette
Vice President, Nuclear Operations,
South Region
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. Abdy Khanpour
Vice President – Engineering Support
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. J. Kammel
Radiological Emergency
Planning Administrator
Division of Emergency Preparedness
Department of Community Affairs
6000 Southeast Tower Drive
Stuart, Florida 34997

Mark E. Warner
Vice President, Nuclear Plant support
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Peter Wells, Acting VP, Nuclear
Training and Performance Improvement
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. Michael Page
Acting Operations Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957-2000

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 152
Renewed License No. NPF-16

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. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B of page 3a to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 152 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

/RA/

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: September 29, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 152

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace Page 3a of Renewed Operating License NPF-16 with the attached Page 3a.

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
1-3	1-3
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 6-27	3/4 6-27
3/4 6-28	3/4 6-28
3/4 6-29	3/4 6-29
3/4 7-20	3/4 7-20
6-15c	6-15c
6-15d	6-15d
6-15e	6-15e
_	6-15f

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

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
ANSI	American National Standards Institute (
ARC	alternate repair criteria
ASME	American Society of Mechanical Engineers
AST	alternative source term
ASTM	American Society for Testing and Materials
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
cpm	count per minute
CR	control room
CRACS	control room air conditioning system
CRE	control room envelope
CRECS	control room emergency cleanup system
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
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
EQ	equipment qualification
°F	degrees Fahrenheit
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
JFD	joint frequency distribution
HEPA	high-efficiency particulate air
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
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
TS	Technical Specification
TSC	technical support center
UFSAR	Updated Final Safety Analysis Report
VFTP	Ventilation System to the Ventilation Filter Testing Program
w/o	weight percent

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By application 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 NPF-16 for St. Lucie Unit 2, 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 supplements dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on June 12, 2008 (73 FR 33460).

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 (GDC) Criterion 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 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.

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 pressurized-water reactor (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 (SGTR) accident, the reactor coolant pump shaft seizure (Locked Rotor) accident (LRA), and the control element assembly (CEA) ejection accident. In addition, the licensee included analyses for the letdown line break accident and the feedwater line break (FWLB) accident, which are not covered in RG 1.183.

The licensee submitted the accident specific input assumptions for each accident as described in the Numerical Applications, Inc. (NAI), "AST Licensing Technical Report for St. Lucie Unit 2," NAI-1 101-044, Revision 2. The inputs and assumptions related to the SGs were based on the replacement SGs, which were scheduled for installation during the fall 2007 refueling outage. These analyses provided for a bounding allowable CR unfiltered air inleakage of 435 cubic feet per minute (cfm). The use of 435 (cfm) as a design-basis value is expected to be above the unfiltered inleakage value to be determined through testing or analysis consistent with the resolution of issues identified in 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 DBA radiological source term analyses bounds the current licensed core thermal power level of 2700 MWt and is, therefore, acceptable to the NRC staff for use in the full implementation of the AST at St. Lucie Unit 2.

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, Atomic Energy Commission, 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 (see the NRC Agencywide Documents Access Management System (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 2.

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, based on the discussion above, 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 license amendment request (LAR).

Regarding post-accident sampling capability, as described in NUREG-0737, Item II.B.3, the licensee cited TS Amendment No. 114, 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 confirms 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 2. 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. Letdown Line Break

8. Feedwater Line Break

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 LPZ and the integrated dose in the St. Lucie Unit 2 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 [emergency core cooling system] 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 updated 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 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 megawatt days per metric ton of uranium (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 performed sensitivity studies to assess the bounding fuel enrichment and bounding burnup values. The assembly source term is based on 102% of rated power or 2754 MWt. The licensee has determined that for rod average burnups in excess of 54,000 MWD/MTU the heat generation rate is limited to 6.3 kw/ft. For non-LOCA events with fuel failures, the licensee applied a bounding radial peaking factor of 1.7 to conservatively simulate the effect of power level differences across the core that might affect the localized fuel failures for assemblies containing the peak fission product inventory.

The licensee used committed effective dose equivalent (CEDE) and effective dose equivalent (EDE) dose conversion factors (DCFs) from Federal Guidance Reports (FGRs) 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 (LOCA)

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 ECCS to provide adequate core cooling which 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.

The licensee considered the following potential DBA LOCA dose contributors to 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 conducted an evaluation of containment sump pH and has 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.

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 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) would 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
- Borated water from the RWT
- Borated water from the safety injection tanks
- Sodium hydroxide solution
- Hydrochloric acid
- Nitric acid

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 calculation, this exception is acceptable.

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 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.

Conclusions

The staff reviewed the analyses and justifications provided by the licensee and concluded that the licensee's proposed actions will maintain sump water basic for 30 days following a LOCA. The post-LOCA water in the RWT, consisting of a residual RWT and the water leaked from the ESF will have a pH lower than 7. However, because of the small volume of this water compared to the total sump water, its effect on molecular iodine generation will be negligible.

3.1.1.2 Assumptions on Transport in the Primary Containment

3.1.1.2.1 Containment Mixing, Natural Deposition and Leak Rate

Section 6.0 of the St. Lucie Unit 2 updated final safety analysis report (UFSAR) describes the containment structure as a steel containment vessel surrounded by a reinforced concrete shield building. The two structures are separated by an annular air space. The containment vessel is a low leakage, cylindrical, steel shell with hemispherical dome and ellipsoidal bottom. The vessel is designed to contain the radioactive material that could be released from a loss of integrity of the reactor coolant pressure boundary. The shield building is a concrete structure that protects the containment vessel from external missiles, provides biological shielding, and provides a means of controlling radioactive fission products that could leak from the containment vessel if an accident should occur.

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 did not credit the removal of organic iodine by natural deposition. 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 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, 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 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 2, the volume of the sprayed region is 2,125,000 ft^3 and the volume of the unsprayed region is 375,000 ft^3. Since the sprayed region represents approximately 85% 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 that 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 2 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 0.01667 hours (60 seconds) after the LOCA, is 6.40 per hour until a decontamination factor (DF) of 50 is reached at 2.65 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 0.01667 hours (60 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 3.06 hours post-LOCA.

The NRC staff has reviewed the licensee's application of credit for iodine removal from the operation of the containment spay system and has found that the analysis follows the applicable regulatory guidance and is, therefore, acceptable.

3.1.1.3 Assumptions on Dual Containments

The St. Lucie Unit 2 UFSAR describes the containment structure as a steel containment vessel surrounded by a reinforced concrete shield building, also referred to as the secondary containment. The two structures are separated by an annular air space. The containment vessel is a low leakage, cylindrical, steel shell with hemispherical dome and ellipsoidal bottom. The vessel is designed to contain the radioactive material that could be released from a loss of integrity of the reactor coolant pressure boundary. The shield building is a concrete structure which protects the containment vessel from external missiles, provides biological shielding, and provides a means of controlling radioactive fission products that could leak from the containment vessel if an accident should occur.

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 SBVS 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 1.28 gallons per hour (gph), representing two times current licensing basis value of 0.64 gph, as specified in RG 1.183, Appendix A, Regulatory Position 5.2. As stated above, actual ECCS leakage would not begin until after the recirculation phase of the accident begins. The licensee assumed that ESF leakage will start at 20 minutes into the event and continue for the 30-day duration of the accident-evaluation period.

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 degrees Fahrenheit (°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.

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.

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

3.1.1.4.2 Assumptions on ESF System Backleakage 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 gallons per minute (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, which becomes elemental, is both a function of the RWT pH and the total iodine concentration. The amount of elemental iodine in the RWT fluid, which then enters the RWT air space, is a function of the temperature-dependent iodine partition coefficient.

The 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 3.7E-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 4.9 to a maximum pH of 5.3 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 0.047.

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 coincident with the beginning 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 2500 cfm released to the environment via the plant vent for a period of 30 seconds with no credit for filtration.

During the time period of 30 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 iodine isotopes from International Commission on Radiological protection 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.

The licensee used conservative assumptions to evaluate the containment purge contribution to the LOCA dose and therefore, the NRC staff finds this evaluation acceptable for the AST LOCA analysis.

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 Cleanup System (CRECS) are required to assure CR habitability. The design of the CRE and overall descriptions of both the CRACS and the CRECS are contained in Sections 6.4 and 9.4.1 of the St. Lucie Unit 2 UFSAR.

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 1000 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 1000 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 30-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, 99% for elemental iodine, and 99% 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.

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 CR. 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 licensee evaluated the contribution to the total dose to the CR operators from direct radiation sources, such as the control room filters, the containment atmosphere, and the released radioactive plume for the LOCA event. The licensee asserts and the NRC staff agrees that the LOCA shine dose contribution is bounding for all other events. 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 asserts that per Table 6.4-2 of the UFSAR, all other sources of direct shine dose to the CR can be considered negligible. 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. 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. 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 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 and is, therefore, acceptable.

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, as well as 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 applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's assumptions are presented in Table 5 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis LOCA will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183, and are therefore acceptable.

3.1.2 Fuel-Handling Accident (FHA)

TID–4844, "Calculations of Distance Factors for Power and Test Reactor Sites" provides the basis for the accident source term used in the current licensed analysis for St. Lucie Unit 2's FHA, as outlined in Chapter 15.7.4.1.2 of the UFSAR. In the CLB for St. Lucie Unit 2, 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 2 reactor core.

For the purpose of implementing AST methodology and support 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. This is a conservative approach considering Section 9.4.2.2.2 of the UFSAR for St. Lucie Unit 2 notes SBVS filtration of releases via the postulated FHA in the FHB. 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, Appendix B, the licensee assumes 100% of the noble gas exits the pool. All fission products released to the environment occurs over a 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 NRC 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 Regulatory Position 1.2 of RG 1.183, Appendix B.

The licensee did not assume an adjustment for high fuel burnup. However, the source term did consider a core average fuel burnup value of 45,000 MWD/MTU with a radial peaking factor to maximize at 1.70. The NRC staff concludes that the licensee’s approach to the accident source term at St. Lucie Unit 2 is acceptable.

3.1.2.2 Transport

Pursuant to guidance provided in RG 1.183, the St. Lucie Unit 2 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 two 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

In order to evaluate the CR habitability for the postulated design basis FHA, the licensee assumed three modes of operation for the CR. The air flow distribution during the normal mode of operation is 1000 cfm of unfiltered fresh air with an assumed value of 500 cfm for unfiltered inleakage. After the radiation monitors activate the emergency signal, both north and south CR intakes are closed simultaneously. This occurs approximately 30 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 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 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 CRECS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 99% for elemental iodine, and 99% for organic iodine.

3.1.2.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated FHA at St. Lucie Unit 2 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 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 licensee's assumptions are presented in Table 6 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that all doses estimated by the licensee for the St. Lucie Unit 2 FHA will 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 (MSLB) 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 15.1 of the St. Lucie Unit 2 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 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.

For the fraction of the core that is assumed to experience fuel centerline melt, the licensee applied the guidance provided in RG 1.183, Appendix H, 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% CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the SGs 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, 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 0.3 gpm total through all SGs and 0.15 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. 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, 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 mass per hr]) 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/cm [grams per cubic centimeters] (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 RG 1.183 and 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 SDC 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 2. 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 which 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 one 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 one 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

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 1000 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 30 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 CRECS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 99% for elemental iodine, and 99% for organic iodine. The CR parameters used in the AST analyses are shown in Table 4 of this SE.

3.1.3.4 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 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 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 licensee’s assumptions are presented in Table 7 and the licensee’s calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee’s methods. The NRC staff finds, with reasonable assurance, that the licensee’s estimates of the dose consequences of a design basis MSLB will 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 (SGTR) 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 15.6.3 of the St. Lucie Unit 2 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 2, 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 2, 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 eight 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 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 2 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 one 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 (approximately 379 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 (PC).

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 2. 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 which does not flash would mix with the bulk water in the SG.

3.1.4.3 CR Ventilation Assumptions for the SGTR

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 1000 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 30-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 409 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 .2.2, “Control Room Atmospheric Dispersion Factors” of this SE. The licensee assumed CRECS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 99% for elemental iodine, and 99% for organic iodine. The CR parameters used in the AST analyses are shown in Table 4 of this SE.

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 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 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 licensee’s assumptions are presented in Table 8 and the licensee’s calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee’s methods. 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 (LRA)

Section 15.3.3 of the UFSAR for St. Lucie Unit 2 describes the LRA. The LRA may be described as an event in which the instantaneous seizure of a single reactor coolant pump (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 a 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 2 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 2 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). In deriving the source term for the subject LRA, St Lucie Unit 2 makes assumptions consistent with regulatory positions illustrated in RG 1.183, Appendix G. Per this guidance, St. Lucie Unit 2 assumes that for the release path analyzed in the event of a LRA, all activity released from the breached fuel assemblies mix 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 fuel 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. Since the St. Lucie Unit 2 analysis assumes that all the fuel meets the burnup limitations in RG 1.183, Footnote 11, the licensee did not apply any adjustment factors for high burnup fuel.

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 2 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 3.2 of RG 1.183.

Additionally, St. Lucie Unit 2 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 2 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 the 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 2 consists of a two-loop RCS, primary and secondary, concurrent 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 occurs 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 2 is currently licensed at a SG accident induced leakage rate of 0.15 gpm per SG and 0.3 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 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 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 into both SGs mixes with the secondary water without flashing. Additionally, in agreement with Regulatory Position 5.5.4, Appendix E, 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

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 1000 cfm. After the radiation monitors activate the emergency signal, both north and south CR intakes are closed simultaneously. This occurs approximately 30 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, the operator acts 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 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 CRECS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 99% for elemental iodine, and 99% for organic iodine.

3.1.5.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated LRA at St. Lucie Unit 2 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 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 licensee's assumptions are presented in Table 9 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the doses estimated by the licensee for the St. Lucie Unit 2 LRA will 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 (CEA) Ejection Accident

Section 15.4.5 of the UFSAR for St. Lucie Unit 2 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 2, the licensees 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 2 assumes in both release scenarios that 9.5% of the fuel rods experience DNB and 0.5% of the fuel will experience fuel centerline melt (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 2 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, the licensee 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 fuel centerline melting and conservatively applied a radial peaking factor of 1.7. Since the St. Lucie Unit 2 analysis assumes that all the fuel meets the burnup limitations in RG 1.183, Footnote 11, the licensee did not apply any adjustment factors for high burnup fuel.

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.

Additionally, St. Lucie Unit 2 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 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 2 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 which 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 2.50E+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 occurs 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 an 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 proposed that the criteria be changed from 0.3 gpm total through all SGs and 0.15 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 technical specifications. TSTF-449, Steam Generator Tube Integrity, changed the SG tube leakage Technical Specification 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

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 1000 cfm. After the radiation monitors activate the emergency signal, both north and south CR intakes are closed simultaneously. This occurs approximately 30 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 CRECS filtration efficiencies, as applied to both the filtered makeup flow and the recirculation flow, of 99% for particulate activity, 99% for elemental iodine, and 99% for organic iodine.

3.1.6.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated CEA accident at St. Lucie Unit 2 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 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 licensee's assumptions are presented in Table 10 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the doses estimated by the licensee for the St. Lucie Unit 2 CEA ejection accident will comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.1.7 Letdown Line Rupture

This event is analyzed as a rupture of a primary coolant letdown line outside of containment. In accordance with the assumptions of UFSAR Section 15.6.2, the dose assessment for this event is based on a double ended rupture of the letdown line in the auxiliary building outside of containment with a direct release to the environment via the plant stack. The licensee evaluated additional releases occurring as a result of secondary side steam relief following the turbine trip and subsequent plant cooldown.

Neither RG 1.183, nor SRP 15.0.1, includes the Letdown Line Rupture event as a DBA. Therefore, the licensee followed the methods employed in SRP, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," with appropriate modifications to maintain consistency with the assumptions in RG 1.183. The licensee used the most restrictive acceptance criteria from SRP 15.0.1, Table 1 and RG 1.183, Table 6, for application in the Letdown Line Rupture event. The approach taken by the licensee, to evaluate the Letdown Line Rupture, using the applicable guidance from SRP, Section 15.6.2, and using the most restrictive acceptance criteria for any of the DBAs considered, is conservative, and therefore, acceptable to the NRC staff.

3.1.7.1 Letdown Line Rupture Source Term

In accordance with SRP 15.6.2, the licensee's analysis assumed an accident-generated or concurrent iodine spike. The RCS activity is initially assumed to be 1.0 µCi/gm DEI as allowed by TS 3.4.8. Iodine is released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. Parameters used by the licensee in the determination of the iodine equilibrium release rate, as well as the iodine activities and the appearance rates, are shown in Table 11 of this SE.

3.1.7.2 Transport

The licensee modeled the Letdown Line Rupture flow rate as 85,788 lbm over 1920 seconds with a flashing fraction of 25.9% as computed using the RG 1.1 83 guidance from Regulatory Position 5.4 of Appendix A for ECCS leakage. All of the activity in the flashed fluid is assumed to be released directly to the environment. The licensee included the dose consequence of the release of additional activity, based on the proposed primary-to-secondary leakage limits, being released via steaming from the ADVs until the RCS is cooled to 212 °F. To evaluate the secondary side releases, the licensee used assumptions consistent with the Locked Rotor accident described in Section 3.1.5 of this SE.

3.1.7.3 CR Ventilation Assumptions for the Letdown Line Rupture

In order to evaluate the CR habitability for the postulated design basis Letdown Line Rupture, the licensee assumed three modes of operation for the CR ventilation system. The licensee used the same CR ventilation assumptions for the Letdown Line Rupture evaluation as was used in the MSLB evaluation described in section 3.1.3.3 of this SE.

3.1.7.4 Conclusion

The licensee evaluated the radiological consequences resulting from the postulated Letdown Line Rupture 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 most restrictive 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 applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's assumptions are presented in Table 11 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis Letdown Line Rupture will comply with the requirements of 10 CFR 50.67 and the most restrictive dose guidelines of RG 1.183, and are therefore acceptable.

3.1.8 Feedwater Line Break (FWLB)

The steam and water release from a postulated feedwater line break results in a loss of secondary coolant which may result in a reactor system cool-down by excessive energy discharge through the break or a reactor system heat-up from the loss of reactor system heat sink. A major feedwater line rupture is defined as a feedwater line break large enough to prevent the addition of sufficient feedwater to the SGs to maintain shell side fluid inventory in the SGs. If the break is postulated in the feedwater line between the isolation valves and the SG, fluid from the SG also is discharged from the break.

The licensee included the rupture of a main feedwater system pipe during plant operation as a part of the AST evaluation. The licensee assumed that the rupture results in the rapid reduction of the secondary water inventory of the affected SG causing a partial loss of secondary heat sink, thereby allowing the heat-up of the RCS. The FWLB is assumed to be located outside of containment resulting in a blowdown of the affected SG to atmosphere from the most limiting release location. The licensee evaluated this event assuming that a LOOP occurs at the time of the trip. Plant cooldown is achieved via the remaining unaffected SG. No fuel failures are postulated to occur as a result of this event.

Neither RG 1.183, nor SRP 15.0.1, includes the FWLB event as a DBA. Therefore, the licensee followed the methods employed in SRP, Section 15.8.2, "Feedwater System Pipe Breaks Inside and Outside Containment (PWR)," with appropriate modifications to maintain consistency with the assumptions in RG 1.183. The licensee used the most restrictive acceptance criteria from SRP 15.0.1, Table 1 and RG 1.183, Table 6, for application in the FWLB event. The approach taken by the licensee to evaluate the FWLB, using the applicable guidance from SRP, Section 15.8.2, and using the most restrictive acceptance criteria for any of the DBAs considered, is conservative, and therefore, acceptable to the NRC staff.

3.1.8.1 FWLB Source Term

The licensee assumed that the initial RCS activity is at the TS limit of 1.0 μCi/gm DEI and 100/E-bar gross activity. In addition, the licensee evaluated the FWLB assuming that the secondary side activity is at the TS 3.7.1.4 limit of 0.1 μCi/gm DEI. The FWLB analysis does not include a coolant spike.

3.1.7.2 Transport

The licensee's analysis assumes that the entire fluid inventory from the faulted SG is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DEI permitted by TS. Additional activity due to primary-to-secondary leakage into the faulted SG is also assumed to be released directly to the environment. Primary-to-secondary leakage is assumed to continue until the affected SG is completely isolated at 12 hours. Primary-to-secondary tube leakage is also postulated to occur in the unaffected SG. The licensee assumed that this activity is diluted by the contents of the SG and released via steaming from the ADVs, along with the initial iodine activity of unaffected SGs. All releases from the unaffected SG continue until the RCS is cooled to 212 °F. To evaluate the secondary side releases, the licensee used assumptions consistent with the Locked Rotor accident described in Section 3.1.5 of this SE.

3.1.8.3 CR Ventilation Assumptions for the FWLB

In order to evaluate the CR habitability for the postulated design basis FWLB, the licensee assumed three modes of operation for the CR ventilation system. The licensee used the same CR ventilation assumptions for the FWLB evaluation as was used in the MSLB evaluation described in section 3.1.3.3 of this SE.

3.1.8.4 Conclusion

The licensee evaluated the radiological consequences resulting from the postulated FWLB 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 most restrictive 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 applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 12 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the dose consequences of a design basis FWLB will comply with the requirements of 10 CFR 50.67 and the most restrictive dose 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 eight limiting DBAs on the CR and offsite EAB and outer boundary of the LPZ 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 2 χ/Q values for the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, Letdown Line Break, and FWLB 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 2 UFSAR.

3.2.1 Meteorological Data

The licensee used six 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 2 used meteorological data for the last six months of 1996, and all of years 1997, 1998, 1999, and 2001. St. Lucie Unit 2 used only the first six months of the year 2000 due to the low recovery rate for the last six 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. These 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 (NOAA) 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 Saint Lucie, FL area. Winds predominantly blew from the east direction at both the lower and upper measurement level during each of the five 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 2 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 NRC staff evaluation and are considered adequate for use in making atmospheric dispersion estimates used in the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, Letdown Line Break, and FWLB dose assessments performed in support of the current LAR for St. Lucie Unit 2.

3.2.2 Control Room Atmospheric Dispersion Factors

The licensee generated new CR χ/Q values for postulated St. Lucie Unit 2 releases for the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, Letdown Line Break, and FWLB 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 NRC 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 2.

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 eight DBAs were assumed to discharge to the environment via nine different source points: the St. Lucie Unit 2 (1) main stack/plant vent (MS), (2) refueling water tank (RWT), (3) closest point of the fuel-handling building, (4) auxiliary building louver (2L-7A), (5) auxiliary building louver (2L-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, 13.1 meters, 11.5 meters, 11.5 meters, 1.6 meters, 14.3 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 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 the adjacent reactor building structure (i.e., 62.9 meters AGL). The primary onsite receptors modeled for the St. Lucie Unit 2 atmospheric dispersion evaluations, as noted in Table 2, were the three St. Lucie Unit 2 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 eight DBAs. These eight 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 1000 cfm for the purpose of analysis). The χ/Q values generated from the release point to the closest CR or least favorable air intake (i.e., north CR intake) are used during this period. Following an accident, the unfiltered inleakage is assumed to continue until 30 seconds from the onset of the accident. At this point, Beta-Gamma scintillation radiation monitors for St. Lucie Unit 2 generate an isolation signal to close both north and south CR intakes simultaneously. The 30-second delay includes 10 seconds for diesel start and 20 seconds for damper actuation with instrument response/signal delay (all times considered conservative measures for the St. Lucie Unit 2 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 30 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 2 uses CRECS for filtration post onset of a DBA. This system is composed of a roughing filter, high-efficiency particulate air (HEPA) prefilter, charcoal adsorbers, and a HEPA after filter and fan. For the purpose of the DBA assessments, credit for dilution of the releases was only given for the main stack/plant vent 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. Accordingly, the licensee notes that the St. Lucie Unit 2 plant vent releases are not in the same 90-degree window as both the north and south CR intakes and the St. Lucie Unit 2 CR is designed with redundant radiation monitors located at both CR intakes. 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 CR 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, Letdown Line Break, and FWLB onsite dose assessments at St. Lucie Unit 2.

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 6 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 1442 meters and outer boundary of LPZ distance of 1490 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 St. Lucie Unit 2 LAR.

3.2.4 Conclusions

The staff reviewed the analyses provided by the licensee regarding the meteorological input, control room χ/Q values, and offsite χ/Q values and concluded that the resulting data are acceptable for use in the LOCA, FHA, MSLB, SGTR, Locked Rotor, CEA Ejection, Letdown Line Break, and FWLB onsite dose assessments at St. Lucie Unit 2. Confirmatory calculations were performed in support of approving the licensee's results. Thus, the NRC staff is confident that the proposed meteorological changes to the St. Lucie Unit 2 CLB will maintain NRC's mission of protecting public health and safety.

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.

3.3.2 The licensee has proposed to revise Surveillance Requirement (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.

3.3.3 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 is modeled after the format of the VFTP in NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants" (Rev. 3.0).

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 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.5 and 4.7.8.1.b.1 since these requirements for in-place HEPA and charcoal adsorber testing are delineated in the referenced RG 1.52, Revision 3.

These testing frequencies specify off-normal as well as normal (i.e., "scheduled") testing and align with the current TS testing frequencies.

The proposed change to delete SRs 4.6.6.1.b.4 and 4.7.8.1.b.4 is an administrative deletion only as these requirements are delineated in the RG 1.52 in-place HEPA and charcoal adsorber testing requirements.

The staff's assessment found these changes acceptable because it provides consistency with the existing format for the CR Emergency Ventilation System filter testing requirements, which is modeled after the format of the VFTP in NUREG-1432, Standard Technical Specifications Combustion Engineering Plants and the guidance of RG 1.52, Revision 3.

3.3.4 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 12% to 9.6%.

The proposed reduction of the acceptance criterion for secondary containment bypass leakage paths via Shield Building Bypass Leakage from 12% to 9.6% increases the safety margin in the plant design, reduces secondary containment bypass leakage, and is therefore acceptable to the NRC staff. The 9.6% value, which is supported by plant leakage test results, increases the allowable CR unfiltered inleakage.

3.3.5 The licensee has proposed to revise the HEPA filter and charcoal adsorber test acceptance criteria for the Shield Building Ventilation. The criteria, as relocated to the VFTP, are revised as follows: The HEPA filter efficiency test acceptance criteria are increased from 99.825% to 99.95%; the in-place charcoal adsorber efficiency test acceptance criteria are increased from 99% to 99.95%; the laboratory charcoal adsorber efficiency test acceptance criterion is increased from 90% to 97.5%.

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 American Society for Testing and Materials (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%.

3.3.6 The licensee has proposed to add the HEPA filter and charcoal adsorber test acceptance criteria for the ECCS Area and Shield Building Ventilation Systems to the VFTP. The criteria, as relocated to the VFTP, are as follows: The HEPA filter efficiency test acceptance criterion remains at the CLB value of 99.95%; the in-place charcoal adsorber efficiency test acceptance test criterion remains at the CLB value of 99.95%; the laboratory charcoal adsorber efficiency test acceptance criterion remains at the CLB value of 97.5%.

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 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 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 engineered safety features (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 American National Standards Institute (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 0.3 gpm through all SGs and 0.15 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 0.3 gpm through all SGs and 0.15 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

Setpoint of automatic actuation of CROAI radiation monitoring instrument is changing from less than or equal to 2x background radiation to less than or equal to 320 count per minute (cpm).

The proposed trip/alarm setpoint of less than or equal to 320 cpm is based on being less than or equal to 8x background radiation. This setpoint has been selected based on one of the two recommendations of Electric Power Research Institute (EPRI) Technical Report TR-102644. This setpoint is low enough to provide the required sensitivity but high enough to avoid nuisance alarms. The licensee has stated that the setpoint will be set between 280 cpm and 320 cpm. The maximum allowable as-found value is 350 cpm with a drift of 12%. If the trip/alarm actuation value exceeds the allowable value of 350 cpm, then the licensee will declare the instrument inoperable and take appropriate actions following the TS and plant corrective action program.

The licensee has conservatively determined the uncertainty value to be 100%. This value was then doubled and a bounding loop uncertainty of 200% was assumed. The method used is consistent with the guidance provided in EPRI's technical report TR-102644. Based on the 200% uncertainty, the upper bound for the setpoint is 960 cpm. The analyzed limiting radiation dose limit based on main steam stop valve stuck open is 2.5 E^{+3} cpm. Considering the worst-case uncertainty of 200%, there is adequate margin between the setpoint and the analyzed value. This meets the guidance of RIS 2006-17.

SR 4.3.3.2 addresses the channel response time. The maximum isolation system actuation time of 50 seconds is conservative, as it takes only 35 seconds for isolation signal actuation and damper closure when the offsite power is available. The licensee verified the time by actual testing. If offsite power is not available, the isolation time is 45 seconds, which includes 10 seconds for emergency diesel generator (EDG) start time. The isolation dampers are on the 0 second EDG load block.

3.5 Modification of Existing Mechanical Equipment

The licensee proposes to revise the licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences for the limiting accidents. The AST analyses were performed for LOCA, FHA, MSLB, SGTR, Reactor Coolant Pump Shaft Seizure (Locked Rotor), and CEA Ejection. Additionally, AST analysis was performed for Inadvertent Opening of a Main Steam Safety Valve for St. Lucie Unit 1. For St. Lucie Unit 2, AST analysis was also performed for Letdown Line Break, and FWLB. These analyses provide for a bounding allowable CR unfiltered air in-leakage of 500 cfm for St. Lucie Unit 1 and 435 cfm for St. Lucie Unit 2. The use of these air in-leakage values as design-basis values was established to be above the unfiltered in-leakage values determined through testing and analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01.

RG 1.183 indicates that evaluations may need to be performed regarding the ability of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses. In an RAI to the licensee dated March 14, 2008, the staff questioned the licensee whether any structural evaluations were performed regarding the adequacy of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses. In its response dated April 14, 2008, the licensee indicated that the isolation function of the containment purge system is performed by two series isolation valves in both the inlet and the exhaust lines. All containment purge isolation valves close automatically on activation of a containment isolation signal. Travel stops have been installed on the purge isolation valves to limit the valves to 40-degrees open (90-degrees being fully-open). This opening was determined in consultation with the valve's supplier, and is such that the critical valve parts will not be damaged by DBA-LOCA loads. In addition, shop tests performed by the valve supplier demonstrated that, since fluid dynamics tend to close a butterfly valve, purge valve closure time during a LOCA was less than or equal to the no-flow time. The isolation valves are Seismic I, safety-related valves. The required isolation time for these valves, as well as for all of the containment isolation valves outlined in UFSAR table, is unchanged by the implementation of the AST. Neither the valves nor the associated ductwork are impacted by, or being modified to support, implementation of the AST. The licensee concluded that the existing structural design and analyses of the down stream ductwork remain valid. After review of the licensee’s response, the staff agrees that the existing structural analyses continue to be applicable after the AST implementation.

The staff also sought additional information regarding any mechanical equipment items and or systems requiring redesign or modification, or whether any new equipment is needed as a result of the implementation of AST term at St. Lucie units. In its response, the licensee indicated that the control circuit for the CR outside air intake radiation monitors is being modified to ensure the circuit fails safe under loss-of-power conditions. As currently designed, the CR outside air intake will not be isolated on a high radiation signal under LOOP conditions. The CR outside air intake radiation monitors are powered from non-essential loads and, as such, are not energized by the EDGs. In addition, under loss of power conditions, the control circuit fails such that it does not initiate the required isolation signal. Prior to implementation of the AST License Amendment, the control circuit for the CR outside air intake radiation monitors will be reconfigured such that it initiates an isolation signal under loss of power conditions. No new equipment will be installed. The only modification will be the reconfiguration of the existing control circuit for the existing radiation monitors. The radiation monitor sample probes are located in the CR outside air intake ducts at a centerline elevation of approximately 79 feet. The radiation monitors are located on the 62 feet elevation of the Reactor Auxiliary Building. The monitors are seismically mounted. Based on a review of this information, the staff finds the licensee's response regarding any redesign or modification of existing mechanical equipment, or need for any new equipment, to be reasonable and acceptable.

Conclusions

Based on a review of the LAR for implementation of AST, and the additional information provided by the licensee for the RAIs, the staff agrees with the conclusion that the existing structural design and analyses of damper and ductwork downstream of the dampers continue to be applicable. The staff also finds the licensee's conclusion, that no existing equipment requires redesign or modification, and that no new equipment is needed as a result of the AST implementation, to be reasonable and acceptable. The staff concurs with the licensee's determination regarding the modification to reconfigure the existing control circuit for the existing radiation monitors.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of an AST at St. Lucie Unit 2. The NRC staff concludes that the licensee used methods of analysis and assumptions consistent with the conservative regulatory requirements and guidance described in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable dose guidelines and criteria referenced in Section 2.0, "Regulatory Evaluation." Based on that comparison, the NRC staff concludes that there is a reasonable assurance that the licensee's estimates of the EAB, LPZ, and CR doses comply with the regulatory requirements. The NRC staff also concludes that there is reasonable assurance that St. Lucie Unit 2, as modified by the requested license amendment, will continue to provide sufficient safety margins and adequate defense-in-depth, under conditions of unanticipated events, and in presence of the uncertainties in accident progression, assumptions, parameters, and analyses outlined above. Therefore, the proposed changes to the licensing basis are acceptable with respect to the radiological consequences of DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the St. Lucie Unit 2 licensing basis is superseded by the revised licensing basis, incorporating the AST as proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE guidelines and criteria of 10 CFR 50.67, or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the St. Lucie Unit 2 design basis, and modified by the present amendment.

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 49579, 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.

Principal Contributors: John Parillo
Natasha Greene
Brian Lee
Krzysztof Parczewski
Chakrapani Basavaraju

Attachment: Tables

Date: September 29, 2008

Table 1
St. Lucie Unit 2 Radiological Consequences Expressed as TEDE [1]
(rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR[4]
LOCA	1.2E+00	2.6E+00	4.5E+00
MSLB – Outside containment (1.8% DNB)	3.3E-01	9.0E-01	4.7E+00
MSLB – Outside containment (0.43% FCM)	3.7E-01	9.8E-01	4.8E+00
MSLB – Inside containment (29% DNB)	5.4E-01	1.1E+00	5.0E+00
MSLB – Inside containment (6.1% FCM)	7.9E-01	1.5E+00	4.9E+00
SGTR Pre-accident spike	2.5E-01	2.4E-01	2.6E+00
Dose acceptance criteria	2.5E+01	2.5E+01	5.0E+00
SGTR Concurrent iodine spike	6E-02	6E-02	6.6E-01
Locked Rotor Accident (13.7% DNB)	2.5E-01	5.6E-01	2.8E+00
FWLB	2E-02	2E-02	8.2E-01
Letdown Line Rupture	3.6E-01	3.6E-01	2.6E+00
Dose acceptance criteria	2.5E+00	2.5E+00	5.0E+00
FHA - Containment	2.9E-01	2.8E-01	8.1E-01
FHA – Fuel Handling Building	2.9E-01	2.8E-01	1.6E+00
CEA Ejection Containment Release[5]	2.6E-01	5.2E-01	2.8E+00
CEA Ejection Secondary Side Release [5]	3.0E-01	6.5E-01	2.9E+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) Allowable unfiltered CR inleakage 500 cfm for all DBAs except MSLB which uses 435 cfm
(5) Assumes 9.5% DNB and 0.5% fuel centerline melt (FCM)

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

Table 2 (Page 1 of 9)

St. Lucie Unit 2
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*	3.79E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.48E-04	4.28E-04	1.99E-04	1.20E-04	9.15E-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	3.32E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.94E-03	1.50E-03	6.47E-04	4.32E-04	3.22E-04

St. Lucie Unit 2
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	3.11E-03	2.73E-03	1.17E-03	8.73E-04	6.76E-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	1.01E-03	8.64E-04	3.72E-04	2.92E-04	2.20E-04

St. Lucie Unit 2
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	1.24E-03	---	---	---	---
After Initiation of Filtered Make-up	Containment Main't Hatch / South CR Intake	8.17E-04	6.08E-04	2.84E-04	1.71E-04	1.29E-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	4.86E-03	---	---	---	---
During CR Isolation	FHB Closest Wall Point / Midpoint CR Intake	3.27E-03	---	---	---	---
After Initiation of Filtered Make-up	FHB Closest Wall Point / South CR Intake	1.86E-03	1.37E-03	6.14E-04	3.90E-04	3.05E-04

Table 2 (Page 4 of 9)

St. Lucie Unit 2
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/m³)				
		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	6.69E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	3.11E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-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/m³)				
		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*	3.79E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.48E-04	4.28E-04	1.99E-04	1.20E-04	9.15E-05

* Credit for dilution was taken in this case.

Table 2 (Page 5 of 9)

St. Lucie Unit 2
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	3.32E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.94E-03	1.50E-03	6.47E-04	4.32E-04	3.22E-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	6.69E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	3.11E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-04

St. Lucie Unit 2
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	6.69E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	3.11E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-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	6.69E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	3.11E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-04

St. Lucie Unit 2
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*	3.79E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.48E-04	4.28E-04	1.99E-04	1.20E-04	9.15E-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	3.32E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.94E-03	1.50E-03	6.47E-04	4.32E-04	3.22E-04

St. Lucie Unit 2
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

O. Letdown Line Break (LLB): Reactor Coolant System (RCS) 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	Stack Vent / North CR Intake*	2.35E-03	---	---	---	---
During CR Isolation	Stack Vent / Midpoint CR Intake*	3.79E-03	---	---	---	---
After Initiation of Filtered Make-up	Stack Vent / South CR Intake*	6.48E-04	4.28E-04	1.99E-04	1.20E-04	9.15E-05

* Credit for dilution was taken in this case.

P. Letdown Line Break (LLB): Steam Generator (SG) 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	6.69E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	3.11E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-04

Table 2 (Page 9 of 9)

St. Lucie Unit 2
Control Room (CR) Atmospheric Dispersion Factors (χ/Q Values)

Q. Feedwater Line Break (FWLB): Intact Steam Generator (SG) 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	6.69E-03	---	---	---	---
During CR Isolation	Closest ADV / Midpoint CR Intake	3.11E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest ADV / South CR Intake	1.88E-03	1.46E-03	5.98E-04	4.23E-04	3.19E-04

R. Feedwater Line Break (FWLB): Affected Steam Generator (SG) 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 FW Line Point / North CR Intake	7.30E-03	---	---	---	---
During CR Isolation	Closest FW Line Point / Midpoint CR Intake	3.32E-03	---	---	---	---
After Initiation of Filtered Make-up	Closest FW Line Point/ South CR Intake	1.94E-03	1.50E-03	6.47E-04	4.32E-04	3.22E-04

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

Table 3

St. Lucie Unit 2
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	1.10E-04	---	---	---	---
	LPZ	1.06E-04	5.91E-05	4.41E-05	2.33E-05	9.32E-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 dose results are expressed to a limit of three significant figures.

Table 4
St. Lucie Unit 2 Control Room Data and Assumptions and Direct Shine Results

Control Room Volume			97,215 ft^3
Normal Operation			
	Filtered Make-up Flow Rate		0 cfm
	Filtered Recirculation Flow Rate		0 cfm
	Unfiltered Make-up Flow Rate		1000 cfm
	Assumed unfiltered inleakage		
		All events except MSLB	500 cfm
		MSLB	435 cfm
Emergency Operation			
Isolation Mode:			
	Filtered Make-up Flow Rate		0 cfm
	Filtered Recirculation Flow Rate		2000 cfm
	Unfiltered Make-up Flow Rate		0 cfm
	Assumed unfiltered inleakage		
		All events except MSLB	500 cfm
		MSLB	435 cfm
Filtered Make-up Mode:			
	Filtered Make-up Flow Rate		450 cfm
	Filtered Recirculation Flow Rate		2000 cfm
	Unfiltered Make-up Flow Rate		0 cfm
	Assumed unfiltered inleakage		
		All events except MSLB	500 cfm
		MSLB	435 cfm
Filter Efficiencies			
	Particulates		99%
	Elemental iodine		99%
	Organic iodine		99%
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 2 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 μCi/gm DEI and 100/E-bar	
Volumetric flow rate due to open purge valves	2500 cfm	
Duration of flow through open purge valves	30 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	2,125,000 ft^3	
Primary containment volume unsprayed region	375,000 ft^3	
Flow rate between sprayed and unsprayed regions	12,500 cfm	
Elemental spray removal coefficients		
0.01667 - 3.06 hours	20 hr^{-1}	
3.06 - 720 hours	0 hr^{-1}	
Particulate spray removal coefficients		
0.016677 - 2.65 hours	6.4 hr^{-1}	
2.65 - 8 hours	0.64 hr^{-1}	
8 - 720 hours	0 hr^{-1}	
Volume of water in containment sump (minimum)	55,739 ft^3	
ECCS Leakage to RAB (2 times allowed limit)	1.28 gallons per hour	
ECCS Flashing fraction		
Calculated	3.4%	
Used for dose determination	10%	
Chemical form of released iodine from ECCS leakage		
Elemental	97%	
Organic	3%	
Particulate	0%	
ECCS area filter efficiencies		
Elemental	95%	
Organic	95%	
Particulate	99%	

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

Initial RWT liquid inventory	52,345 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	4.900
9.72	4.909
22.22	4.921
97.22	4.987
720.00	5.319
Time dependent RWT total iodine concentration (gm-atom/liter)	
Selected times in hours	RWT Iodine concentration
0.00	0.00E+00
9.72	1.25E−06
22.22	2.84E−06
97.22	1.08E−05
720.00	3.70E−05
Time dependent RWT liquid temperature	
Selected times in hours	Temperature (°F)
0.00	100.0
9.72	100.0
22.22	100.0
97.22	102.5
720.00	103.8
Time dependent RWT elemental iodine fraction	
Selected times in hours	Elemental iodine fraction
0.00	0.00E+00
9.72	6.22E−03
22.22	1.33E−02
97.22	3.71E−02
720.00	3.24E−02
Time dependent RWT partition coefficient (PC)	
Selected times in hours	Elemental iodine PC
0.00	45.65
9.72	45.65
22.22	45.65
97.22	43.52
720.00	42.46

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

LOCA Adjusted iodine release rate from RWT

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

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

Transport assumptions

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

Table 6
St. Lucie Unit 2 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
Minimum pool water depth	23 feet
Fuel clad damage gap release fractions	
I-131	8%
Remainder of halogens	5%
Kr-85	10%
Remainder of noble gases	5%
Alkali metals	12% (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	
Isolation time	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	500 cfm

Table 7 (Page 1 of 2)

St. Lucie Unit 2 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 µCi/gm DEI and 100/E-
Secondary coolant iodine activity	0.1 µCi/gm DEI
Primary to secondary leak rates	0.25 gpm per SG
RCS density based on RCS conditions	5.9 – 7.84 lbm/gallon
Time to terminate SG tube leakage	12 hours
Reactor coolant system (RCS) mass	423,700 lbm (minimum)
SG secondary side mass	
Intact SG	105,000 lbm (minimum mass)
Faulted SG	260,000 lbm (maximum mass)
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	8250
0.25 – 0.50	4382
0.50 – 0.75	4915
0.75 – 1.0	5600
1.0 – 1.5	5250
1.5 – 2.25	4934
2.25 – 4.0	4098
4.0 – 8.0	2671
8.0 – 10.32	3247
10.32 – 720	0
SG secondary side iodine partition coefficients	
Intact SG	100
Faulted SG	1(none)
Chemical form of iodine released from the secondary side	
Elemental	97%
Organic	3%
Particulate	0%

Table 7 (Page 2 of 2)
St. Lucie Unit 2 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	2.50E+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
Credit for containment sprays	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	
Isolation time	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	435 cfm

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 2 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	423,700 lbm
Concurrent iodine spike	452,000 lbm
Secondary coolant system mass	
Minimum for SG tube leakage	105,000 lbm
Maximum for secondary side release	260,000 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	78,040	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	32.47 lbm/min

St. Lucie Unit 2 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	164.8	79,124
I-132	92.0	44,146
I-133	239.3	114,868
I-134	111.6	53,559
I-135	161.1	77,310

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

I-131	48.8
I-132	10.2
I-133	60.778
I-134	6.07
I-135	30.3

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

Isolation time (total)	409.2 seconds
Start of release from ADVs	379.2 seconds
Delay for DG start, fan start and dampers	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	500 cfm

Table 9
St. Lucie Unit 2 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%
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 during SG tube uncovery	5%
Time to reach 212 ˚F terminating steam release	10.32 hours
RCS mass – minimum used to maximize dose	423,700 lbm
SG secondary side mass	
Minimum for SG leakage	105,000 lbm/SG
Maximum for secondary side release	260,000 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	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

Control room ventilation assumptions

Isolation time	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	500 cfm

St. Lucie Unit 2 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
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 Rx trip	1 hour
Flashing fraction during SG tube uncovery	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	423,700 lbm
SG secondary side mass	
Minimum for SG leakage	105,000 lbm/SG
Maximum for secondary side release /	260,000 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	
Isolation time	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	500 cfm

St. Lucie Unit 2 Data and Assumptions for the CEA Ejection Accident

Containment volume	2.50E+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	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

Table 11 (Page 1 of 2)
St. Lucie Unit 2 Data and Assumptions for the Letdown Line Rupture Accident

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
Iodine spike appearance rate	500 times equilibrium rate
Duration of accident initiated spike	8 hours
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 Rx 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	
For RCS activity	423,700 lbm
Concurrent iodine spike	452,000 lbm
SG secondary side mass	
Minimum for SG leakage	105,000 lbm/SG
Maximum for secondary side release	260,000 lbm/SG
Letdown line rupture flow rate	85,788 lbm over 1920 seconds
Letdown line flashing fraction	25.9%

Letdown line rupture steam release rates (lbm/min) for time period (hrs)

Time (Hours)	SG release rate (lbm/min)
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 (Page 2 of 2)
St. Lucie Unit 2 Data and Assumptions for the Letdown Line Rupture Accident

Iodine equilibrium appearance assumptions	Value
Maximum letdown flow rate	150 gpm at 120 ˚F, 650 psia 0%
Maximum identified RCS leakage	10 gpm
Maximum unidentified RCS leakage	1 gpm

RCS Iodine Inventory (Ci) for 8-hr concurrent spike with an appearance rate factor of 500

Isotope	Appearance rate (Ci/min)		8 hour total (Ci)
	@ 1 μCi/gm DEI	With concurrent spike	
I-131	0.4920	246	118,077
I-132	0.2745	137	65,8880
I-133	0.7144	357	171,445
I-134	0.3330	167	79,928
I-135	0.4807	240	115368

Chemical form of iodine released	
Particulate	0%
Elemental	97%
Organic	3%

Control room ventilation assumptions	
Isolation time	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	500 cfm

Table 12
St. Lucie Unit 2 Data and Assumptions for the Feedwater Line Break Accident

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
SG tube leakage	0.25 gpm per SG
Time to terminate SG tube leakage	12 hours
RCS density based on RCS conditions	5.9 – 7.8 lbm/gallon
Time to recover SG tubes following Rx 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
Unaffected SG secondary side mass	
Minimum for SG leakage	105,000 lbm/SG
Chemical form of iodine released	
Particulate	0%
Elemental	97%
Organic	3%
Control room ventilation assumptions	
Isolation time	30 seconds
Filtered makeup flow time	1.5 hours
Assumed unfiltered inleakage	500 cfm

Feedwater Line Break steam release rates (lbm/min) for time period (hrs)

Time (Hours)	Unaffected SG release rate (lbm/min)
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