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

June 23, 2011

Mr. Mano Nazar
Executive Vice President and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING ALTERNATIVE SOURCE TERM (TAC NOS. ME1624 AND ME1625)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 244 to Renewed Facility Operating License No. DPR-31 and Amendment No. 240 to Renewed Facility Operating License No. DPR-41 for the Turkey Point Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Licenses and the Technical Specifications in response to your application dated June 25, 2009, as supplemented by letters dated July 21, July 30, August 26, 2009, February 10, March 15, April 14, April 28, May 21, June 11, June 23, June 25, September 2, September 15, October 13, December 14, 2010, and May 11, 2011.

The amendments revise the Turkey Point, Units 3 and 4 licensing bases to adopt the alternative source term as allowed in Title 10 of the *Code of Federal Regulations*, Section 50.67. The licensee revised the plant-licensing basis through reanalysis of the following radiological consequences of the Updated Final Safety Analysis Report, Chapter 14 Accidents: Loss-of-Coolant Accident, Main Steam Line Break, Steam Generator Tube Rupture, Locked Rotor, Rod Cluster Control Assembly Ejection, Fuel Handling Accident, Spent Fuel Cask Drop, and Waste Gas Decay Tank Rupture.

Section 4.0 and Section 5.0 of the Safety Evaluation, Enclosure 3 contain a list of licensee commitments and license conditions, respectively. The license conditions are to be implemented before the implementation of this license amendment.

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,

Jason C. Paige, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 244 to DPR-31
2. Amendment No. 240 to DPR-41
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 244
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated June 25, 2009, as supplemented by letters dated July 21, July 30, August 26, 2009, February 10, March 15, April 14, April 28, May 21, June 11, June 23, June 25, September 2, September 15, October 13, December 14, 2010, and May 11, 2011, 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, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-31 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 244 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Further, Renewed Facility Operating License No. DPR-31 is amended to add new license conditions, 3.H.1., 3.H.2., and 3.H.3. which read:

   1. FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR Part 50 Appendix A, GDC 19. The new intakes will be located near the ground level extending out from the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes. In addition, FPL will provide to the NRC a confirmatory assessment which demonstrates that the requirements of 10 CFR 50 Appendix A, GDC 19 will be met. The confirmatory assessment will follow the methodology in Amendment 244 [the alternative source term amendment] including the methods used for the establishment of the atmospheric dispersion factors (X/Q values).

   2. FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

   3. The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey Point, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

4. This license amendment is effective as of its date of issuance and shall be implemented by the completion of the Cycle 26 refueling outage for Unit 3.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: June 23, 2011

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 240
Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated June 25, 2009, as supplemented by letters dated July 21, July 30, August 26, 2009, February 10, March 15, April 14, April 28, May 21, June 11, June 23, June 25, September 2, September 15, October 13, December 14, 2010, and May 11, 2011, 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, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-41 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 240 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Further, Renewed Facility Operating License No. DPR-41 is amended to add new license conditions, 3.I.1., 3.I.2., and 3.I.3. which read:

   1. FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR Part 50 Appendix A, GDC 19. The new intakes will be located near the ground level extending out from the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes. In addition, FPL will provide to the NRC a confirmatory assessment which demonstrates that the requirements of 10 CFR 50 Appendix A, GDC 19 will be met. The confirmatory assessment will follow the methodology in Amendment 240 [the alternative source term amendment] including the methods used for the establishment of the atmospheric dispersion factors (X/Q values).

   2. FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

   3. The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey Point, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

4. This license amendment is effective as of its date of issuance and shall be implemented by the completion of the Cycle 27 refueling outage for Unit 4.

FOR THE NUCLEAR REGULATORY COMMISSION

N Wa[illegible] for

Douglas A. Broaddus, 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: June 23, 2011

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 244 RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 240 RENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace Page 3 of Renewed Operating License DPR-31 with the attached Page 3.
Replace Page 5 of Renewed Operating License DPR-31 with the attached Page 5.
Add the attached Page 6 to Renewed Operating License DPR-31.

Replace Page 3 of Renewed Operating License DPR-41 with the attached Page 3.
Replace Page 6 of Renewed Operating License DPR-41 with the attached Page 6.
Add the attached Page 7 to Renewed Operating License DPR-41.

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

Remove pages	Insert pages
ix	ix
1-3	1-3
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 7-16	3/4 7-16
3/4 7-16a	3/4 7-16a
3/4 7-16b	3/4 7-16b
3/4 7-17	3/4 7-17
3/4 7-25	3/4 7-25
6-17	6-17
6-18a	6-18a
6-19	6-19

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2300 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 244 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than July 19, 2012.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

H. Alternative Source Term (AST) Modifications

1. FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19. The new intakes will be located near the ground level extending out from the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes. In addition, FPL will provide to the NRC a confirmatory assessment which demonstrates that the requirements of 10 CFR 50 Appendix A, GDC 19 will be met. The confirmatory assessment will follow the methodology in Amendment 244 [the alternative source term amendment] including the methods used for the establishment of the atmospheric dispersion factors (X/Q values).

2. FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

3. The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey Point, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight July 19, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Signed by
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications for Unit 3
Appendix B – Environmental Protection Plan

Date of Issuance: June 6, 2002

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2300 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 240 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than April 10, 2013.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(d) FPL will not move any fuel assemblies into the Unit 4 SFP subsequent to the successful completion of startup physics tests for Unit 4 Cycle 25.

I. Alternative Source Term (AST) Modifications

1. FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19. The new intakes will be located near the ground level extending out from the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes. In addition, FPL will provide to the NRC a confirmatory assessment which demonstrates that the requirements of 10 CFR 50 Appendix A, GDC 19 will be met. The confirmatory assessment will follow the methodology in Amendment 240 [the alternative source term amendment] including the methods used for the establishment of the atmospheric dispersion factors (X/Q values).

2. FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

3. The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey Point will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight April 10, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Signed by
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications for Unit 4
Appendix B – Environmental Protection Plan

Date of Issuance: June 6, 2002

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

FREQUENCY NOTATION

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE Equivalent I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE - 133

1.13 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

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

GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

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

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

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.25 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133.

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

ACTION:

a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT 1-131, verify DOSE EQUIVALENT 1-131 is less than or equal to 60 microcuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT 1-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT 1-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT 1-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT 1-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

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TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. NOT USED		
2. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131	a) 1 per 14 days. b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.	1, 2, 3, 4
4. Radiochemical Isotopic Determination Including Gaseous Activity	Monthly	1, 2, 3, 4
5. Isotopic Analysis for DOSE EQUIVALENT XE-133	1 per 7 days	1, 2, 3, 4
6. NOT USED		

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3/4.6.2.3 RECIRCULATION pH CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Recirculation pH Control System shall be OPERABLE.

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

ACTION:

With the Recirculation pH Control System inoperable, restore the buffering agent to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 72 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The Recirculation pH Control System shall be demonstrated OPERABLE:

a. At least once per 18 months by

   1. Verifying that the buffering agent baskets are in place and intact;

   2. Collectively contain $\geq$ 11061 pounds (227 cubic feet) of sodium tetraborate decahydrate, or equivalent.

CONTAINMENT SYSTEMS

3/4.6.3 Not Used

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 The Control Room Emergency Ventilation System shall be OPERABLE with:

a. Three air handling units,

b. Two of three condensing units,

c. Two control room recirculation fans,

d. Two recirculation dampers,

e. One filter train,

f. Two isolation dampers in the normal outside air intake duct,

g. Two isolation dampers in the emergency outside air intake duct,

h. Two isolation dampers in the kitchen area exhaust duct, and

i. Two isolation dampers in the toilet area exhaust duct.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

a.1 With one air handling unit inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable air handling unit to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.2 With two condensing units inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore at least one of the inoperable condensing units to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.3 With one recirculation fan inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable fan to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (continued)

a.4 With one recirculation damper inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the recirculation dampers in the open position and place the system in recirculation mode** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.5 With the filter train inoperable, e.g., an inoperable filter, and/or two inoperable recirculation fans, and/or two inoperable recirculation dampers, immediately suspend all movement of irradiated fuel, and, immediately, initiate action to implement mitigating actions, and, within 24 hours, verify mitigating actions ensure control room occupant radiological exposures will not exceed limits and, within 7 days, restore the filter train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.6 With an inoperable damper in the normal outside air intake, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the normal outside air intake isolation dampers in the closed position and place the system in recirculation mode** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.7 With an inoperable damper in the emergency outside air intake, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the emergency outside air intake isolation dampers in the open position** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.8 With an isolation damper inoperable in the kitchen area exhaust duct, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or isolate the flow path** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.9 With an isolation damper inoperable in the toilet area exhaust duct, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or isolate the flow path** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

**If action is taken such that indefinite operation is permitted and the system is placed in recirculation mode, then movement of irradiated fuel may resume.

LIMITING CONDITION FOR OPERATION (continued)

MODES 5 and 6:

With the Control Room Emergency Ventilation System inoperable, suspend all operations involving CORE ALTERATIONS, movement of fuel in the spent fuel pool, or positive reactivity changes. This ACTION shall apply to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F;

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes***;

c. At least once per 18 months or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system, or (4) after complete or partial replacement of a filter bank by:

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% DOP and halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%***.

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 - 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement***, and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration***.

d. At least once per 12 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%***;

e. At least once per 18 months by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

f. At least once per 18 months by verifying operability of the kitchen and toilet area exhaust dampers.

***As the mitigation actions of TS 3.7.5 Action a.5 may include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c and d.

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.7.9 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (DOSE EQUIVALENT Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the Reactor Coolant System total activity exceeds the limit of Specification 3.4.8.

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Deleted

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following deviations or exemptions:

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment interval pressure for the design basis loss of coolant accident, P_a, is 49.9 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding 1.0 L_a, the As-left leakage rate acceptance criterion is $\leq 0.75\ L_a$, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

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

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

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.

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

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

The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

1. For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection, tubes with service-induced flaws located greater than 17.28 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 17.28 inches below the top of the tubesheet shall be plugged upon detection.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1 .1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

Reports required on an annual basis shall include:

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 0.25 microcurie per gram DOSE EQUIVALENT I-131.

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 244 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND

AMENDMENT NO. 240 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT PLANT, UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated June 25, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092050277), as supplemented by letters dated July 21 (ADAMS Accession No. ML100680718), July 30 (ADAMS Accession No. ML092230254), August 26, 2009 (ADAMS Accession No. ML092520357), February 10 (ADAMS Accession No. ML100470318), March 15 (ADAMS Accession No. ML100770223), April 14 (ADAMS Accession No. ML101090027), April 28 (ADAMS Accession No. ML101200063), May 21 (ADAMS Accession No. ML101450028), June 11 (ADAMS Accession No. ML101650648), June 23 (ADAMS Accession No. ML101760019), June 25 (ADAMS Accession No. ML101800222), September 2 (ADAMS Accession No. ML102510127), September 15 (ADAMS Accession No. ML102630160), October 13 (ADAMS Accession No. ML102880046), December 14, 2010 (ADAMS Accession No. ML103500344), and May 11, 2011 (ADAMS Accession No.ML11137A080), Florida Power and Light (FPL, the licensee) proposed an amendment to the Technical Specifications (TSs) for Turkey Point Plant, Units 3 and 4. The requested changes propose to adopt the alternative source term (AST) as allowed in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67. The changes affect License Conditions of Renewed Operating Licenses DPR-31 and DPR-41 for Units 3 and 4, respectively.

The application provides the TS changes and evaluations of the radiological consequences of design-basis accidents (DBAs) for implementation of a full-scope AST pursuant to 10 CFR 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 July 21, July 30, August 26, 2009, February 10, March 15, April 14, April 28, June 11, June 23, June 25, September 2, September 15, October 13, December 14, 2010, and May 11, 2011, 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 December 29, 2009 (74 FR 68870).

The supplement dated May 21, 2010, changed the scope of the application as originally noticed. Due to the changes, the application was renoticed and published in the *Federal Register* on July 13, 2010 (75 FR 39978).

2.0 REGULATORY EVALUATION

2.1 Background

In the early 1970s, the U.S. Nuclear Regulatory Commission (NRC) staff issued regulatory guidance for evaluating the consequences of DBAs using the radiological source term described in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Since the publication of TID-14844, significant advances in understanding timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents have occurred. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 used updated research to provide more realistic estimates of the accident source term that were physically based on, and could be applied to, the design of future light-water power reactors. In addition, the NRC determined that the analytical approach, based on the TID-14844 source term, would continue to be adequate to protect public health and safety for the current licensed power reactors. The NRC staff also determined that current licensees may wish to use the NUREG-1465 source term, referred to as the AST, in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and RG 1.183 (July 2000). Issuance of RG 1.183 provided the first comprehensive guidance for analyzing DBAs for radiological consequences using the AST.

A holder of an operating license issued prior to January 10, 1997 (Operating Reactors), or a holder of a renewed license under 10 CFR Part 54, "Conditions of licenses," whose initial operating license was issued prior to January 10, 1997, can in accordance with 10 CFR 50.67, voluntarily revise the accident source term used in design basis radiological consequence analyses. However, to ensure proper implementation of the AST, the NRC required, in 10 CFR 50.67(b), that "A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under Section 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report."

Turkey Point was licensed prior to the 1971 publication of Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50. As such, Turkey Point is not licensed to the current GDC of 10 CFR Part 50 Appendix A. Section 1.3 of the Turkey Point Updated Final Safety Analysis Report (UFSAR) provides a summary of the 1967 GDC proposed by the U.S. Atomic Energy Commission (AEC) as amended by the Atomic Industrial Forum (circa October 2, 1967). The licensee indicates throughout the Turkey Point UFSAR that it is committed to continued compliance with the proposed GDC to which it was licensed in 1967. A review of the 1967 proposed GDC shows that the proposed GDC applicable to the structures, systems, and components (SSCs) credited in the Turkey Point AST are comparable to the requirements of the current GDC provided in 10 CFR Part 50 Appendix A.

2.2 Accident Dose and Ventilation

In addition to developing the AST and providing regulatory guidance for its implementation, the NRC determined that new dose criteria for protection of public health and safety were appropriate and included these performance-based criteria in 10 CFR 50.67. Paragraph (b)(2)(i) of 10 CFR 50.67 states 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 0.25 sievert (Sv) or 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE). Paragraph (b)(2)(ii) of 10 CFR 50.67 states that an individual located at any point on the outer boundary of the low population zone, 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 0.25 Sv (25 rem) TEDE.

In addition to providing regulatory dose criteria for protection of the public, the NRC requires that control room (CR) personnel be protected from the potential radiological consequences of a DBA. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the design criteria for water-cooled nuclear power plants. GDC-19, "Control Room," states that adequate radiation protection shall be provided to permit access and occupancy of the CR under accident conditions without personnel receiving radiation exposures in excess of 5 rem (0.05 Sv) whole body, or its equivalent to any part of the body, for the duration of the accident.

For those plants applying to use the AST, paragraph (b)(2)(iii) of 10 CFR 50.67 provides CR habitability criteria similar to GDC-19 and states that adequate radiation protection is provided to permit access to and occupancy of the CR under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

A design basis radiological consequence analysis is intended to be based upon a major accident, or possible event, resulting in dose consequences not exceeded by those from any accident considered credible. The analysis is sometimes referred to as the Maximum Hypothetical Accident (MHA) or the maximum credible accident. Unlike the design basis loss of coolant accident (LOCA), used to evaluate the emergency core cooling system (ECCS) requirements of 10 CFR 50.46, the general scenario used to postulate a maximum hypothetical dose consequence accident does not represent any specific accident sequence. Rather, the MHA is intended to be a surrogate to enable a deterministic evaluation of the response of a facility's engineered safety features (ESFs) such as the primary containment system. Although the maximum hypothetical dose consequence large break LOCA (LBLOCA) is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of possible occurrence including other dose consequence DBAs such as the fuel-handling accident (FHA). These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, airborne activity product transport, and atmospheric dispersion.

RG 1.183 states in Regulatory Position 1.2 that a complete implementation of an AST would upgrade all existing radiological analyses. Although a complete reassessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses for operating reactors would generally not be necessary. Full implementation is a modification of the facility design basis that addresses all characteristics of the AST: composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the facility's licensing basis to specify the AST in place of the previous accident source term and establishes new TEDE dose acceptance criteria. This applies not only to the analyses performed in the application, which may only include a subset of the plant analyses, but also to all future design basis analyses. As a minimum for full implementation, the maximum credible dose consequence LBLOCA (MHA) must be determined and then analyzed using the guidance in Appendix A of RG 1.183.

As stated in Regulatory Position 5.2 of RG 1.183, the DBAs addressed in the appendices of RG 1.183, other than Appendix A for LOCA dose consequence analysis where the source term is defined by regulation, were selected from accidents that may involve damage to irradiated fuel. The inclusion or exclusion of a particular dose consequence 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.

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, RGs, and standards:

- 10 CFR 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," Rev. 0, February 1972.
- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 3, June 2001.
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, November 1982.
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000.
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Rev. 0, June 2003.
- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Rev. 0, May 2003.
- NUREG-0409, "Iodine Behavior in a PWR [Pressurized-Water Reactor] Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985.

- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric 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," Branch Technical Position (BTP) 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak Or Failure," Rev. 3, March 2007.
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000.
- NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," 1988.
- NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992.
- USNRC, Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms," March 7, 2006.
- Technical Specification Task Force TSTF-490, "Deletion of E Bar Definition and Revision to RCS [reactor coolant system] Specific Activity Tech Spec."

2.3 Structural Integrity

Appendix A to 10 CFR Part 100 requires that SSCs necessary to assure the capability of the plant to mitigate the consequences of accidents, which could result in exposures comparable to the guideline exposures provided in 10 CFR Part 100, be designed to remain functional during and after a safe shutdown earthquake (SSE). The NRC staff's review in the area of mechanical and civil engineering focuses primarily on the structural integrity, including seismic qualification and seismic interactions, of SSCs such as the Control Room Emergency Ventilation System (CREVS), sodium tetraborate decahydrate (NaTB) baskets and the normal containment cooler (NCC) coils which are credited in the implementation of the AST at Turkey Point.

The NRC staff's evaluation considered 10 CFR 50.55a, GDC 1 and GDC 2. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of affected SSCs under postulated accident conditions, as analyzed with the implementation of an AST at Turkey Point. The acceptance criteria are based on the continued conformance with the requirements of: 10 CFR 50.55a, and GDC 1, as they relate to SSCs being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; and to GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of loadings imposed on these SSCs due to the occurrence of extraordinary natural phenomena, such as earthquakes and tornadoes, combined with the effects of accident conditions.

The guidance associated with the implementation of an AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." With respect to the mechanical and civil engineering aspects of the AST implementation, RG 1.183 indicates that licensees must evaluate nonradiological impacts on a facility which are a consequence of the implementation of an AST methodology. For this particular AST license amendment request (LAR), the licensee is requesting to implement a full scope AST at Turkey Point. As described in RG 1.183, a full scope AST implementation refers to the licensee's request to recalculate the dose consequences of select DBAs to address all five characteristics of the AST (i.e., the composition, magnitude, chemical and physical forms of the radioactive material and the timing of the material's release). Additional guidance for the review can also be found in Section 15.0.1 of the SRP (or NUREG-0800).

2.4 Electrical Systems

Appendix A of 10 CFR Part 50, GDC 17, "Electric Power Systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of SSCs that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

Appendix A of 10 CFR Part 50, GDC 18, "Inspection and Testing of Electric Power Systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

As required by 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," the safety related electrical equipment that are relied upon to remain functional during and following design basis events be qualified for accident (harsh) environments. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

As required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," preventative maintenance activities must not reduce the overall availability of the systems, structures, or components.

RG 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design-Basis Accidents

The licensee performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183, and SRP Section 15.0.1. The licensee performed AST analyses for the

PWR DBAs identified in RG 1.183 that could potentially result in significant control room (CR) and offsite doses. These include the LOCA, the FHA, the main steam line break (MSLB) accident, the steam generator tube rupture accident (SGTR), the reactor coolant pump shaft seizure (Locked Rotor) accident, the rod cluster control assembly (RCCA) ejection accident, the waste gas decay tank (WGDT) Rupture, and the Spent Fuel Cask Drop (SFCD) Accident.

RG 1.183 does not provide specific guidance for the evaluation of the WGDT Rupture or the Spent Fuel Cask Drop. For the WGDT Rupture event, the licensee has adopted the guidance from BTP 11-5 of the SRP and from Regulatory Information Summary (RIS) 2006-04. RIS 2006-04 establishes the staff's acceptance criteria for the AST dose limit of 100 mrem TEDE at the exclusion area boundary (EAB), which corresponds numerically to the annual radiation dose limit for individual members of the public as specified in 10 CFR Part 20, "Standards for Protection Against Radiation." The licensee evaluated the EAB dose over the duration of the event rather than for the worst 2-hour period. In addition, for the evaluation of the CR, the licensee applied a limit of 5 rem TEDE maintaining consistency with all AST CR evaluations. The NRC staff finds that the licensee's application of dose criteria for the WGDT Rupture event to be consistent with established staff positions and regulatory guidance.

For the Spent Fuel Cask Drop event, the licensee has proposed to use the FHA dose limits in RG 1.183 that reflect the guidance of SRP 15.7.5 "Spent Fuel Cask Drop Accidents." SRP 15.7.5 defines acceptable radiological consequences for a postulated SFCD accident as being well within the exposure guideline values of 10 CFR 100.11. SRP 15.7.5 defines well within as 25 percent or less of the 10 CFR Part 100 exposure guideline values. When translated into the TEDE dose criterion used to evaluate dose consequences using the AST, the dose acceptance criterion for the Cask Drop Accident becomes 6.3 rem TEDE, which is the same as the off-site FHA dose limit in RG 1.183. In addition, for the evaluation of the CR, the licensee applied a limit of 5 rem TEDE maintaining consistency with all AST CR evaluations.

The licensee submitted the accident specific input assumptions for each accident as described in the Numerical Applications, Inc. (NAI), "AST Licensing Technical Report for Turkey Point Units 3 and 4," NAI-1396-045, Revision 2. These analyses provide for a bounding allowable CR unfiltered air inleakage of 100 cubic feet per minute (cfm). The use of 100 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, "Control Room Habitability Assessment Guidance," and Generic Letter 2003-01, "Control Room Habitability." CR inleakage testing performed at Turkey Point in 2003 indicates less than 10 cfm of unfiltered inleakage providing a significant margin between the bounding dose analysis inleakage value of 100 cfm and the measured CR unfiltered inleakage.

The DBA radiological source term used in the AST analyses was developed based on a core power level of 2652 megawatts thermal (MWt). The core power level used in the AST analysis of 2652 MWt represents a core power level of 2644 MWt with a 0.3% increase to account for measurement uncertainties. The use of 2652 MWt for the AST DBA radiological source term analyses bounds the current licensed core thermal power level of 2300 MWt.

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 (AEC), 1962, "Calculation of Distance Factors for Power and Test Reactors Sites," accident source term will remain the licensing basis for equipment environmental qualification (EQ).

Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memo dated April 30, 2001 (ADAMS Accession No. ML011210348) and in the June 2001 NUREG-0933, Supplement 25 (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 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 staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ considerations.*

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."

The licensee cited the resolution of Generic Issue 187 as described for the justification of maintaining the current licensing basis (CLB) source term for EQ, and for the radiological dose analyses for post-accident vital area access as described in NUREG-0737, Item II.B.2. The licensee asserts, and the NRC staff concurs that since the calculated post-accident vital area access dose rates are not expected to be significantly impacted by the AST during the first 30-days following a LOCA, the conclusions of the shielding study would not change significantly by expressing the mission dose in terms of TEDE.

A full implementation of the AST is proposed for Turkey Point Units 3 and 4. 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 (LOCA)

2. Fuel-Handling Accident (FHA)

3. Main Steamline Break Accident (MSLB)

4. Steam Generator Tube Rupture Accident (SGTR)

5. Reactor Coolant Pump Shaft Seizure Accident (Locked Rotor)

6. Rod Cluster Control Assembly (RCCA) ejection accident

7. Waste Gas Decay Tank (WGDT) Rupture

8. Spent Fuel Cask Drop (SFCD)

The DBA dose consequence analyses evaluated the integrated TEDE dose at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low population zone (LPZ) and the integrated dose in the Turkey Point Units 3 and 4, 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. The 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 (attachment to this SE).

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

In accordance with RG 1.183, Regulatory Position 3.1, the licensee generated the core and worst case fuel assembly radionuclide inventories for use in determining source term inventories using the ORIGEN code version 2.1. The licensee assumed a period of irradiation that was sufficient to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. For the LOCA, in which all 157 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 a bounding average assembly power of 16.892 MWt. The AST power level is based on the expected extended power uprate (EPU) power level of 2644 MWt with an additional 0.3% to account for calorimetric uncertainty yielding an AST power level of 2652 MWt. The licensee performed sensitivity studies to assess the bounding fuel enrichment and bounding burnup values. The minimum fuel enrichment is based on an historical minimum of 3.0 weight percent (w/o) and a bounding maximum assumed fuel enrichment of 5.0 w/o. The license used the limiting isotopic concentration from either enrichment to determine the core inventory for use in the dose consequence analyses. The licensee conservatively assumed that a maximum assembly uranium mass of 463 kilograms applies to all 157 fuel assemblies. The use of ORIGEN is in accordance with RG 1.183 guidance.

The gap release fractions specified in RG 1.183 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.

For the Fuel Handling and SFCD events, the licensee modified the gap fractions specified in Table 3 of the RG 1.183 to account for high burnup fuel using the guidance from NUREG/CR-5009. The use of guidance from NUREG/CR-5009 results in gap fractions that are approximately twice as large as the gap fractions specified in Table 3 of RG 1.183. The licensee conservatively applied these larger gap fractions to all of the rods in the fuel assembly even though only a small percentage of the rods may exceed the burnup limits specified in RG 1.183. The use of guidance from NUREG/CR-5009 is acceptable to the NRC staff and has been accepted in other AST submittals.

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 DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance.

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 that results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analyses.

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of the primary RCS piping, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. Consistent with the guidance in RG 1.183, 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 dose consequence LOCA for a PWR, RG 1.183 specifies 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 design basis dose consequence LOCA.

In the evaluation of the design basis dose consequence LOCA analysis, the licensee considered dose contributions from the following potential activity release pathways:

- Containment leakage,
- Engineered safety feature (ESF) system leakage into the Auxiliary Building,
- ESF system backleakage into the refueling water tank (RWST), and
- Containment purge at event initiation.

The licensee considered the following potential DBA LOCA dose contributors to the occupants of the CR:

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

3.1.1.1 LOCA Source Term

After a LOCA, a variety of different chemical species are released from the damaged core. One of them is radioactive iodine. This iodine, when released to the outside environment, will significantly contribute to radiation doses. It is, therefore, essential to keep it confined within the plant's containment. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plant," iodine is released from the core in three different chemical forms; at least 95% is released in ionic form as cesium iodide (CsI) and the remaining 5% as elemental iodine (I_2) and hydriodic acid (HI); the release contains at least 1% of each I_2 and HI.

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 potential of Hydrogen (pH) is maintained at or above 7.0.

CsI and HI are ionized in water and are, therefore, soluble. However, elemental iodine is scarcely soluble. To sequester the iodine in water, it is desirable to maintain as much as possible of the released iodine in ionic form. In radiation environments existing in containment, some of the ionic iodine dissolved in water is converted into elemental form. The degree of conversion varies significantly with the pH of water. At a higher pH, conversion to elemental form is lower and at pH greater than 7.0 it becomes negligibly small. The relationship between the rate of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950, "Iodine Evolution and pH Control."

Turkey Point Units 3 and 4 use sodium tetraborate to control post-LOCA sump pH. Currently, the post-LOCA sump pH is controlled by manually adding sodium tetraborate through the Chemical Volume and Control System in accordance with Emergency Operating Procedures. The proposed license amendment includes a new TS and a commitment to install a passive system to control the post-LOCA sump pH. The proposed passive system would consist of 10 stainless steel wire mesh baskets filled with sodium tetraborate. The baskets will be located in the lower regions of the containment such that they are fully submerged in sump fluid prior to the onset of sump recirculation flow. The licensee performed an analysis showing that within approximately 45 minutes of the beginning of a postulated LOCA, enough sodium tetraborate is dissolved to maintain a pH greater than 7.0. This analysis used conservative assumptions for sump fluid temperature which provide a slower dissolution rate than would be expected in the actual event. In addition, the sodium tetraborate in the baskets is modeled as a solid block and only the outer surface area is considered to be actively dissolving. No credit is taken for enhanced dissolution due to flow through the basket, and no credit is taken for crumbling of the sodium tetraborate that would create a greater surface area exposed to the sump fluid.

The licensee provided information regarding the assumptions and calculations used to verify that the sump pH would remain greater than 7.0 following a LOCA. The licensee's analysis considered minimum and maximum boron concentrations and volumes for the refueling water storage tank, accumulator, and reactor coolant system. Additional inputs included the impact of strong acids generated by radiation of cable insulation and sump water.

Strong acids generated from cable insulation and sump fluids under accident radiation levels were inputs to the licensee's pH calculation. Radiolysis of chloride-bearing cable insulation results in the generation of hydrochloric acid. The licensee used data from plant walkdowns to determine the quantity of cable material in containment. The bounding quantity of cable material was found in Unit 3 and was determined to be 41,742 pounds (lb). The licensee assumed that the entire mass of cable insulation was chloride-bearing and that it was all subject to radiolysis. This is noted as a conservatism in the calculation since many cables will have non-chlorinated insulation material. In addition, many cables have chlorinated jackets only, with nonchlorinated conductor insulation. The licensee determined that 1.08E+02 g-mol of hydrochloric acid are produced at the onset of containment spray recirculation and 5.86E+03 g-mol of hydrochloric acid are produced at 30 days after the LOCA. Radiolysis of the sump water results in the generation of nitric acid. The licensee determined that 4.77E-01 g-mol of nitric acid are produced at the beginning of containment spray recirculation and 4.54E+01 g-mol of nitric acid are generated at 30 days. The staff finds the licensee's calculations for strong acid generation acceptable.

The licensee conducted an evaluation of containment sump pH and has determined that the sump pH will be maintained at or above 7.0. 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 RG 1.183 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.

3.1.1.2 Assumptions on Transport in the Primary Containment

3.1.1.2.1 Containment Mixing, Natural Deposition and Leak Rate

Section 5.1.1 of the Turkey Point, Units 3 and 4, Final Safety Analysis Report (FSAR) describes the principal design basis for the containment structure as being capable of withstanding, without loss of integrity, the peak pressure resulting from any size pipe break including the MHA. The MHA is defined as the release of the water in the system through a double-ended break of a reactor coolant pipe, coincident with a loss of normal off-site power.

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 using the methodology from SRP 6.5.2. The licensee credited an elemental iodine natural deposition removal coefficient of 5.58 hr^{-1}, which was applicable to both the sprayed and unsprayed volume of the containment. The licensee did not credit the removal of organic iodine by natural deposition.

The licensee credited a natural deposition removal coefficient of 0.1 hr^{-1} for all aerosols in the unsprayed region of containment, and for all aerosols in the sprayed region when the sprays are not operating.

RG 1.183, 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 0.2% per day for the first 24 hours based on the proposed TS 6.8.4h. In accordance with the guidance in RG 1.183, after 24 hours the licensee reduced the assumed containment leak rate to 0.1% per day for the duration of the accident.

3.1.1.2.2 Containment Spray Assumptions

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

RG 1.183, Appendix A, Regulatory Position 3.3 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."

For Turkey Point Units 3 and 4, the volume of the sprayed region is 534,442 ft^3 out of a total containment volume of 1,550,000 ft^3. Since the sprayed region represents approximately 34.5% of the total containment volume, the Turkey Point containment building atmosphere is not considered to be a single, well-mixed volume. The licensee modeled the containment spray removal by dividing the containment into three separate regions; a sprayed region, an unsprayed region above the operating deck and an unsprayed region below the operating deck. The licensee determined the most conservative mixing rates between the regions based on a sensitivity study evaluating various combinations of containment fans and sprays. Mixing flow rates up to 375,000 cfm between lower and upper unsprayed regions and 990,000 cfm between upper sprayed and unsprayed regions were chosen to conservatively cover the possible combinations of sprays and emergency fans that may be available during an accident scenario. The licensee also noted its results are consistent with NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings."

> *The mixing rates for the containment sprayed and unsprayed regions are based on a GOTHIC analysis which produced results consistent with NUREG/CR-4102, 'Air Currents Driven by Sprays in Reactor Containment Buildings.' Precedent for similar containment mixing results is established in the Fort Calhoun Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 198 to DPR-40 issued April 4, 2001.*
>
> *The GOTHIC analysis utilized for Turkey Point to demonstrate the level of spray induced mixing in containment included both subdivided and lumped parameter models. The detailed subdivided models were used to calculate flow patterns produced by the containment sprays and the emergency containment coolers. Gas concentrations from the subdivided models were compared with concentrations in the lumped parameter model and used to determine equivalent mixing flow rates for the lumped model.*

Using the guidance from SRP 6.5.2, the licensee determined that the aerosol removal rate from the effects of the containment spray system is 6.44 per hour until a decontamination factor (DF) of 50 is reached at greater than 3.06 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, which is consistent with the guidance in RG 1.183. Using the guidance from 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 just over 2.305 hours post-LOCA.

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

3.1.1.3 Assumptions on ESF System Leakage

The ESF system leakage results from the operation of the ECCS. ESF leakage develops when ECCS systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. 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 throughout 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 ESF 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.

For the LOCA analysis of ESF leakage, the licensee used a leakage value of 4650 cubic centimeters per hour (cc/hr), representing two times current licensing basis value of 2325 cc/hr, as specified in RG 1.183, Appendix A, Item 5.2. As stated above, actual ECCS leakage would not begin until after the recirculation phase of the accident begins. The license assumed that ESF leakage will start at the earliest time the recirculation flow occurs in these systems and continue for the 30-day duration.

3.1.1.3.1 Assumptions on ESF System Leakage to the auxiliary building

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

The licensee calculated the fractional iodine release or flashing fraction for ESF leakage as 9.2%. However, the licensee used a flashing fraction of 10%, as prescribed in RG 1.183, for conservatism. As discussed in Section 3.1.1.1 of this SE, the licensee has determined that the pH of the containment sump will not fall below 7.0 for the duration of the accident. Therefore, any iodine present in the containment sump is expected to remain in solution.

The licensee assumed that the ECCS leakage is released directly into the auxiliary building and released instantaneously into the environment with no credit for auxiliary building ECCS area filtration. The licensee did not credit a reduction of activity released to the auxiliary building as a result of dilution or holdup. In accordance with RG 1.183, for ESF leakage into the auxiliary building, 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 assumptions for ECCS leakage to the auxiliary building and has determined that the analysis follows the guidance of RG 1.183 and is, therefore, conservative.

3.1.1.3.2 Assumptions on ESF System Backleakage to the Refueling Water Tank (RWST)

The licensee evaluated the dose consequence from ECCS backleakage to the RWST by assuming an initial backleakage rate of 0.1 gallons per hour (gph). The licensee based the analyzed backleakage rate of 0.1 gph on doubling of the expected total seat leakage through both sets of motor operated valves which isolate the recirculation flow from the RWST. The licensee assumed that this leakage starts at 15 minutes into the event when recirculation begins and continues throughout the 30-day analysis period. The licensee notes that based on the leakage rate and the size of the piping, the leakage would not be expected to reach the RWST for an extended period of time after recirculation begins. Conservatively, the licensee did not credit the realistic time period for determining when the leakage reaches the RWST. Instead the licensee conservatively assumed that the backleakage reaches the RWST instantaneously, not crediting radioactive decay during the expected transit time. The licensee did credit the expected transit time for the determination of the temperature of the leakage reaching the RWST. The licensee further notes that based on the small leak rate and pipe length between the ECCS isolation valves and the RWST, ECCS fluid backleakage is not likely to reach the RWST during the first 30 days of the event.

Based on sump pH remaining at 7.0 or above, the iodine in the sump solution is assumed to all be nonvolatile. However, when introduced into the acidic solution of the RWST inventory, there is a potential for the particulate iodine to convert into the elemental form. The fraction of the total iodine in the RWST which becomes elemental is both a function of the RWST pH and the total iodine concentration. The amount of elemental iodine in the RWST fluid which then enters the RWST 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 RWST 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 8.03×10^{-8} gm-atom per liter at 30 days. For conservatism, the licensee assumed a constant value of 1.0×10^{-7} gm-atom per liter for the analysis.

Because of the small backleakage rate, the licensee conservatively assumed that RWST pH remains constant at the initial value of 3.0 for the duration of the analysis period. Using a constant and conservative value for RWST pH and the total iodine concentration in the RWST 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 RWST elemental iodine fraction will range from 0.0 at the beginning of the event to a maximum of 0.0882.

The licensee assumed that the elemental iodine in the liquid region of the RWST will become volatile and partition between the liquid and vapor space in the RWST based upon the partition coefficient for elemental iodine as described in NUREG/CR-5950. Because the RWST 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 RWST, the licensee calculated the flow rate of the released activity from the vapor space within the RWST based upon the displacement of air by the incoming backleakage. The licensee calculated the elemental iodine release rate from the RWST by multiplying the displacement air flow rate times the elemental iodine concentration in the RWST vapor space. The licensee used the same approach to evaluate the organic iodine release rate from the RWST. 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 guidance of RG 1.183, the licensee assumed that the particulate portion of the leakage is retained in the liquid phase of the RWST. Therefore, the total iodine release rate is the sum of the elemental and organic iodine release rates. The licensee modeled the activity from ECCS components and from RWST leakage as an unfiltered ground level releases from the location of the RWST. For the ECCS leakage, the licensee has determined that the atmospheric dispersion factors from the RWST to the CR emergency intakes are more limiting than from any of the Auxiliary Building penetrations.

The NRC staff has reviewed the licensee's analysis assumptions for ECCS backleakage into the RWST and has determined that the analysis follows conservative engineering assumptions and applicable regulatory guidance in RG 1.183 and NUREG/CR-5950, and is therefore conservative.

3.1.1.4 Assumptions on Containment Purging

The licensee evaluated the radiological effects of a containment purge 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. Consistent with the current licensing basis, the containment purge contribution is modeled as a volumetric flow rate of 7000 cfm released to the environment for a period of 8 seconds before containment isolation valves end the release. As stated in the UFSAR the maximum closure time for these valves is 5 seconds from receipt of an automatic isolation signal. The containment purge is conservatively modeled as a ground level release via the plant vent with no credit for filtration.

For a 30-second time period following onset of the accident, 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 beginning 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 on the proposed RCS TS radionuclide concentrations of 0.25 micro curie per gram (μCi/gm) dose equivalent I-131 (DEI) and 447.7 μCi/gm dose equivalent Xe-133 (DEX). The licensee's current TS definition of DEI references the DCFs for individual iodine isotopes from International Commission on Radiological Protection document ICRP 30, which are equivalent to the rounded committed dose equivalent (CDE) thyroid values from FGR 11 for iodine isotopes. With the approval of this LAR, the licensee will change the TS definition of DEI to reference the CDE thyroid values from FGR 11 for iodine isotopes.

The licensee used assumptions consistent with RG 1.183 to evaluate the containment purge contribution to the LOCA dose and, therefore, the NRC staff finds this evaluation is conservative for the AST LOCA analysis.

3.1.1.5 CR Habitability

3.1.1.5.1 CR Ventilation Assumptions for the LOCA

The CR ventilation system as described in Section 9.9 of the Turkey Point, Units 3 and 4, UFSAR, provides assurance of CR habitability during normal operating conditions, anticipated operational occurrences, and design basis accident conditions. For the LOCA analysis, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. After the start of the event, the CR will be isolated on either a safety injection signal, or a high radiation signal from either the containment or the normal CR intake. The licensee applied a 30-second delay to account for the time required to reach the signal, the time to start the diesel generator and the time for damper actuation. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

3.1.1.5.2 CR Direct Shine Dose Assumptions

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

- 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, and
- Direct shine from radioactive material in systems and components inside or external to the CR envelope including radioactive material buildup on the CR ventilation filters.

RG 1.196 defines the CR envelope (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 determined the direct shine dose from four different sources to the CR operator after a postulated LOCA event. These sources are the containment building, the CR recirculation filters, the external cloud that envelops the CR and the containment purge duct penetration. 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 in the containment. By using these outputs the licensee established a bounding CR filter inventory 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 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 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 assumptions and results of the licensee's CR direct shine dose evaluation are presented in Table 4 of this SE. The remaining LOCA assumptions are presented in Table 5 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the staff determined 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)

The FHA, as described in Section 14.2.1.2 of the UFSAR, consists of the drop of a single fuel assembly in the fuel-handling building (FHB) or inside containment. The UFSAR description of the FHA specifies that all of the fuel rods in a single fuel bundle are damaged as a result of being dropped during fuel handling. In addition, a minimum water level of 23 ft is maintained above the damaged fuel assembly for both the containment and FHB release locations. The FHA is assumed to occur at the minimum time allowable for fuel movement as specified in TS limiting condition for operation (LCO) 3.9.3, which is 72 hours. In accordance with RG 1.183, the licensee assumed that the release to the environment from the FHA occurs over a 2-hour period.

3.1.2.1 FHA Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. 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.

The licensee modified the gap fractions specified in Table 3 of the RG 1.183 to account for high burnup fuel using the guidance from NUREG/CR-5009. The gap fractions from NUREG/CR-5009 as used by the licensee are approximately twice those specified in RG 1.183 and are conservative for use in the FHA analysis.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP) depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3 the licensee assumed that the chemical form of radioiodine released from the fuel consists of 95% CsI, 4.85% I_2, and 0.15% organic iodine. The CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This results in a final iodine distribution in the pool water of 99.85% I_2 and 0.15% organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

3.1.2.2 FHA Transport

Pursuant to guidance provided in RG 1.183, the licensee assumed that all of the fission products released from the reactor cavity or SFP are released to the environment over a 2-hour period. The licensee conservatively modeled the release to the environment as a ground-level release for all scenarios considered. 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.

When the correction to the elemental iodine decontamination factor as discussed in Item 8 of RIS 2006-04 (ADAMS Accession No. ML053460347) is applied, 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). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 70% elemental and 30% organic species.

The licensee assumed that the minimum allowable water depth of 23 ft covers the underlying damaged fuel assembly in both the reactor cavity and SFP for the FHA analyzed in the subject LAR. The damaged fuel rods are assumed to release 100% of the gap activity to the surrounding water. The water provides a decontamination of the released iodine activity 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 gap activity is assumed to be released 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 two modes of operation for the CR ventilation. These assumptions are consistent with actual CR ventilation system operation for each mode of operation. For the FHA in containment, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. After the start of the event, the CR will be isolated on a high radiation signal from the containment monitor. The licensee applied a 30-second delay to account for the time required to reach the signal, the time to start the diesel generator and the time for damper actuation. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow.

For the FHA in the FHB, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. The CR is assumed to be manually isolated by operator action 30 minutes after the initiating event. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow.

The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

3.1.2.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated FHA at Turkey Point, Units 3 and 4, 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 assumptions used in the analysis are presented in Table 6 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the staff determined that the doses estimated by the licensee for the Turkey Point Units 3 and 4, 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, which leads to an uncontrolled release of steam. The resultant depressurization of the steam system causes the main steam safety valves (MSSVs) 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. For the MSLB outside containment, the affected steam generator (SG), hereafter referred to as the faulted SG, rapidly depressurizes and releases the initial contents of the SG to the environment. Because the LOOP renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

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 MSLB analysis conservatively 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.

The licensee evaluated the radiological consequences of an MSLB outside containment that bound those of an MSLB inside containment. The MSLB accident is described in Section 14.2.5.2 of the Turkey Point Units 3 and 4, UFSAR. RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for a PWR MSLB.

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 released activity should be the maximum coolant activity allowed by TSs including the effects of pre-accident and concurrent iodine spiking. The licensee's evaluation indicates that no fuel damage would be expected to occur as a result of a MSLB 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 the TSs. In addition, 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 MSLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TSs for a spiking condition. The maximum iodine concentration allowed by TSs 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. The proposed RCS TS limit for normal operation is 0.25 μCi/gm DEI. In accordance with RG 1.183, 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 500 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.

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% I_2, and 0.15% 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.

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 LCO of 0.1 µCi/gm DEI.

3.1.3.2 MSLB Transport

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 SGs. The analysis assumes that activity is released as reactor coolant enters the SGs due to primary-to-secondary leakage. All noble gases associated with this leakage are assumed to be released directly to the environment.

The licensee assumed a primary-to-secondary leak rate of 0.6 gpm total for all SGs with a maximum of 0.2 gpm to any one SG. This is in accordance with proposed change to the accident induced leakage performance criteria of the SG Program as described in proposed TS Section 6.8.4.j.b.2. The licensee has proposed that the criteria be changed from the current value of 1 gpm total through all SGs and 500 gallons per day (gpd)(0.35 gpm) through any one SG, to a total of 0.6 gpm through all SGs and 0.2 gpm through any one SG. This proposed change continues to maintain margin to the operational leakage limit specified in the TSs.

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) should be consistent with the basis of the parameter being converted. The alternate repair criteria (ARC) leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3)." Accordingly the licensee converted the volumetric leak rate to mass leak rates using a density of 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

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 is in operation and releases from the SGs have been terminated." In accordance with RG 1.183, the licensee assumed that the primary-to-secondary leakage continues until the temperature of the RCS is cooled to less than 212°F. The licensee determined that this would occur at 125.4 hours after the initiating event. The licensee assumed that the release of radioactivity from the unaffected SGs would continue until residual heat removal (RHR) is capable of removing decay heat and for providing for any further cooldown. The licensee determined that this would occur at 63 hours after the initiating event.

In the letter dated April 14, 2010, the licensee provided additional information describing the defense in depth considerations supporting the cool down assumptions used in the MSLB dose consequence analysis. The licensee asserts that the current licensing and design basis for Turkey Point credits the safety-related MSSVs in response to DBAs in order to prevent over-pressurization and that the CR operators utilize the safety-related Atmospheric Dump Valves (ADVs) to depressurize and cool down the RCS to the RHR cut-in temperature and pressure. The licensee further stated that these ADVs are safety related but their upgraded digital controls (e.g., hand switches and controllers) are quality related since the components support control functions, including safe shutdown. The valves are provided with two redundant air sources; instrument air and a backup nitrogen gas system. The nitrogen system serves as an alternate power source to the ADVs in the event of loss of all AC power. Furthermore, for additional defense in depth considerations, operators may operate the valves by installing a temporary gas supply. The licensee noted that for analytical considerations, the time to place RHR in service was conservatively assumed to be 63 hours and that further analysis has since determined that it will actually only take 25.5 hours. The NRC staff finds that the combination of safety-related valves, redundant air sources as well as backup procedures for manual operator actions supports cool down time assumptions that are conservative for use in the dose consequence analyses.

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 with no mitigation. For the unaffected SGs that are 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 system 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 SGs is limited by the moisture carryover from the SGs." Accordingly, the licensee assumed that the radioactivity in the bulk water of the unaffected SGs 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 SGs.

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 SGs for up to 30 minutes following a reactor trip. During this period, the licensee assumed that the fraction of primary-to secondary leakage that flashes to vapor would rise through the bulk water of the SG into the steam space and be immediately released to the environment with no mitigation. The licensee calculated a flashing fraction of 11% based on the thermodynamic conditions in the reactor and secondary coolant. The licensee assumed that the leakage that does not flash would mix with the bulk water in the SG.

In a letter dated April 14, 2010, the licensee provided additional information describing the basis for the assumption of a 30-minute period of SG tube bundle uncovery. The licensee stated that the Turkey Point Units 3 and 4, operating history from rapid power transients from 100% to 0% indicates that 30 minutes may be assumed as a conservatively long tube uncovery period. Operating history shows that following unplanned reactor trips from 100% power, SG level can drop below the indicating range of the narrow range level instrument for several minutes. Data collected for several of these events includes the SG narrow range level profile. In each of these cases, level is restored to the indicating range in under 10 minutes. In addition, auxiliary feedwater will automatically initiate to maintain SG level. For these reasons, the licensee asserts and the NRC staff agrees that an assumed value of 30 minutes to tube recovery is considered to be a conservative value for use in the MSLB dose consequence analysis.

The licensee assumed that releases from the faulted main steam line would occur from the main steam line associated with the most limiting atmospheric dispersion factors. To evaluate releases from the unaffected SGs, the licensee conservatively modeled the release from the MSSV or ADV with the most limiting atmospheric dispersion factors.

3.1.3.3 CR Ventilation Assumptions for the MSLB

The CR ventilation system as described in Section 9.9 of the Turkey Point, Units 3 and 4 UFSAR, provides assurance of CR habitability during normal operating conditions, anticipated operational occurrences, and design basis accident conditions. For the MSLB analysis, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. After the start of the event, the CR will be isolated on a safety injection signal. The licensee applied a 41.5-second delay for CR isolation, which is comprised of an 11.5-second delay to account for the initiation of the safety injection signal and a 30-second delay to account for the signal processing and damper actuation. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

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 assumptions used in the analysis are presented in Table 7 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the staff determined 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 an SGTR accident as a part of the full implementation of an AST. The postulated SGTR event is described in Section 15.6.3 of the Turkey Point, Units 3 and 4, 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 resulting in an automatic reactor trip. 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 dose 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 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 TSs. 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 TSs for a spiking condition. The maximum iodine concentration allowed by TSs 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. The proposed TS RCS limit for normal operation is 0.25 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.

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 SGTR 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 assumed a primary-to-secondary leak rate of 0.6 gpm total for all SGs with a maximum of 0.2 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 proposed TS Section 6.8.4.j.b.2. The licensee has proposed that the criteria be changed from the current value of 1 gpm total through all SGs and 500 gpd (0.35 gpm) through any one SG, to a total of 0.6 gpm through all SGs and 0.2 gpm (288 gpd) through any one SG. The operational limit on primary to secondary leakage limit in any one SG is 150 gpd at room temperature. Therefore this proposed change continues to maintain approximately a 2-to-1 margin to the operational leakage limit specified in the TS.

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 SGTR. 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) should be consistent with the basis of the parameter being converted. The alternate repair criteria (ARC) leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3)." Accordingly the licensee converted the volumetric leak rate to mass leak rates using a density of 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

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 SGs should be assumed to continue until shutdown cooling is in operation and releases from the SGs have been terminated." Consistent with the CLB, the activity release from 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 SGs.

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 SGs 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 SGs within 30 minutes 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. The staff finds the licensee's approach with respect to scrubbing to be conservative.

During the first 291 seconds (0.808 hours) 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. During this time period the licensee assumed a partition factor of 100 for releases through the condenser pathway. After 291 seconds, the condenser is no longer available due to the assumed LOOP.

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

In accordance with applicable regulatory guidance, the licensee assumed a partition coefficient of 100 for iodine. The licensee also 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 SGs for up to 30 minutes following a reactor trip. During this period, the licensee assumed that the fraction of primary-to secondary leakage that flashes to vapor would rise through the bulk water of the SG into the steam space and be immediately released to the environment or the containment with no mitigation. The licensee calculated a flashing fraction of 11% based on the thermodynamic conditions in the reactor and secondary coolant. The licensee assumed that the leakage that does not flash would mix with the bulk water in the SG.

3.1.4.3 CR Ventilation Assumptions for the SGTR

The CR ventilation system, as described in Section 9.9 of the Turkey Point, Units 3 and 4, UFSAR, provides assurance of CR habitability during normal operating conditions, anticipated operational occurrences, and DBA conditions. For the SGTR analysis, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. After the start of the event, the CR will be isolated on a safety injection signal that causes a plant trip at 291 seconds. The licensee applied a 30-second delay to account for the signal processing and damper actuation. After CR isolation is complete at 321 seconds, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

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 assumptions used in the analysis are presented in Table 8 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the staff determined 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 14.1.9 of the UFSAR for Turkey Point Units 3 and 4, describes the LRA. The LRA may be described as an event in which the instantaneous seizure of a single reactor coolant pump (RCP) rotor 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 loop is rapidly reduced. 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 departure from nucleate boiling (DNB) margin, and SG tube leakage due to a significant pressure differential between the primary and secondary systems. The fission products from the damaged fuel are assumed to mix instantaneously and homogeneously in the primary coolant. Primary coolant activity transfers to the secondary system via SG tube leakage. A portion of the primary coolant activity from SG tube leakage together with secondary activity is postulated to be released to the environment via the ADVs and MSSVs.

3.1.5.1 LRA Source Term

The licensee estimated the amount of fuel damage caused by the LRA by determining the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel cladding barrier is breached. The licensee determined that for the LRA, there would not be any fuel melt and that 15% of the fuel rods would exceed the criterion for DNB resulting in breached cladding and the release of gap activity into the RCS. The licensee incorporated the guidance from Table 3 of RG 1.183, which specifies the noble gas, alkali metal, and iodine fuel gap release fractions for the 15% of the fuel rods that experience breached cladding. 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 1.65.

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% I_2, 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 specification 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.

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 G, the licensee has determined that the LRA will result in a limited amount of fuel clad damage. Specifically, the licensee's LRA analysis assumes that 15% of the total of fuel rods 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.

In a letter dated April 14, 2010, the licensee provided additional information describing the basis for the use of the fuel gap release fractions as specified in Table 3 of RG 1.183. The licensee stated that the fraction of rods in the core reaching the DNB limit in the LRA is assessed for each reload cycle. As part of the reload analysis, it is confirmed that rods exceeding the DNB limit for the LRA do not have a rod average linear heat generation rate greater than 6.3 kw/ft if the rod has a burnup greater than 54 GWD/MTU and, therefore, meet the restrictions specified in footnote 11 of RG 1.183. Therefore, the staff determined that the gap fractions used in the LRA dose consequence analysis are conservative and therefore acceptable.

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. The licensee evaluated the RCS and secondary activity in the LRA calculations and assumed that the initial RCS activity is at the proposed TS limit of 0.25 μCi/gm DEI and 447.7 DEX. In addition, the licensee assumed that the initial secondary activity is at the TS limit of 0.1 μCi/gm DEI.

3.1.5.2 LRA Transport

Pursuant to guidance provided in RG 1.183, Appendix G, the licensee analyzed the primary-to-secondary release path, with subsequent releases from the secondary system to the atmosphere via steaming. This analysis is based on the assumption that all of the fission products released from the damaged fuel cladding are instantaneously and homogeneously mixed throughout the primary coolant. 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. Conservatively, the licensee used ground-level release assumptions for the secondary release scenario of the LRA.

The licensee assumed a primary-to-secondary leak rate of 0.6 gpm total for all SGs with a maximum of 0.2 gpm to any one SG. This is in accordance with proposed change to the accident induced leakage performance criteria of the SG Program as described in proposed TS Section 6.8.4.j.b.2. The licensee has proposed that the criteria be changed from the current value of 1 gpm total through all SGs and 500 gpd (0.35 gpm) through any one SG, to a total of 0.6 gpm through all SGs and 0.2 gpm through any one SG. The operational limit on primary to secondary leakage limit in any one SG is 150 gpd at room temperature. Therefore this proposed change continues to maintain approximately a 2-to-1 margin to the operational leakage limit specified in the TSs. Using the applicable regulatory guidance from RG 1.183, the licensee converted the volumetric leak rate to mass leak rates using a density of 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

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 SGs should be assumed to continue until shutdown cooling is in operation and releases from the SGs have been terminated." In accordance with RG 1.183, the licensee assumed that the primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F. The licensee determined that this would occur at 125.4 hours after the initiating event. The licensee assumed that radioactivity is released to the atmosphere via steaming from the SG ADVs and MSSVs until the RHR system is capable of removing decay heat. The licensee has determined that this would occur at 63 hours after the initiating event.

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 a portion of the postulated leakage is assumed to flash to vapor based on the thermodynamic conditions in the RCS and secondary coolant system immediately following plant trip when tube uncovery is postulated. The licensee assumed that the portion of leakage that immediately flashes to vapor rises through the SG bulk water into the steam space and is released without credit for any reduction due to scrubbing within the SG bulk water. The licensee assumed that the primary-to-secondary leakage would mix with the SG bulk water without flashing during periods of total tube submergence.

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 SGs is limited by the moisture carryover from the SGs." 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 30 minutes following a reactor trip. During this period, the licensee assumed that the fraction of primary-to-secondary leakage that flashes to vapor would rise through the bulk water of the SG into the steam space and be immediately released to the environment or the containment with no mitigation. The licensee calculated a flashing fraction of 11% based on the thermodynamic conditions in the reactor and secondary coolant. The licensee also assumed that the leakage that does not flash would mix with the bulk water in the SG.

The licensee conservatively assumed that releases from the SGs would occur from the MSSV or ADV with the most limiting atmospheric dispersion factors.

3.1.5.3 CR Ventilation Assumptions for the LRA

The CR ventilation system as described in Section 9.9 of the Turkey Point, Units 3 and 4, UFSAR, provides assurance of CR habitability during normal operating conditions, anticipated operational occurrences, and DBA conditions. For the LRA analysis, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. After the start of the event, the CR will be isolated on a high radiation signal from the normal CR intake monitors. The licensee applied a 60-second delay to account for the time to reach the setpoint, signal processing and damper closure time. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

3.1.5.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated LRA at Turkey Point, Units 3 and 4, 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 assumptions used in the analysis are presented in Table 9 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the staff determined that the doses estimated by the licensee for the Turkey Point Units 3 and 4, 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 Rod Cluster Control Assembly (RCCA) Ejection Accident

Section 14.2.6 of the UFSAR for Turkey Point, Units 3 and 4, describes the RCCA ejection accident as the mechanical failure of a RCCA and drive shaft resulting in the ejection of a single RCCA and drive shaft from the reactor core. The primary consequence of the described mechanical failure is a rapid reactivity insertion together with an adverse core power distribution leading to a reactor trip and possible fuel rod damage. In accordance with RG 1.183, the licensee evaluated two independent release cases in the event of an RCCA ejection accident.

The first case 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 case assumes that all of the activity released from the damaged fuel is fully dispersed in the primary coolant system 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 RCCA ejection event using the accident source term pursuant to guidance provided in RG 1.183, Appendix H. The licensee 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 RCCA ejection accident.

3.1.6.1 RCCA Ejection Accident Source Term

For both cases analyzed, the licensee assumed that 10% of the fuel rods experience DNB and 0.25% of the fuel will experience fuel centerline melt (FCM) as a result of the RCCA ejection from the reactor core. 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 1.65. In accordance with the guidance from RG 1.183, Appendix H, the licensee made the following assumptions for the two cases analyzed:

For case 1, the containment leakage release pathway, it is assumed that in the event of an RCCA ejection 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. In accordance with RG 1.183, 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. The licensee credits effective controls to limit the pH in the containment sump to 7.0 or higher. Therefore, 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% I_2, and 0.15% organic iodide.

For case 2, the secondary system release pathway, it is assumed that in the event of a RCCA ejection 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 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. Additionally, the licensee assumed that the initial equilibrium secondary activity is at the TS limit of 0.1 μCi/gm DEI.

3.1.6.2 RCCA Transport

Pursuant to guidance provided in RG 1.183, Appendix H, the licensee evaluated the RCCA ejection accident for two separate transport 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 second case assumes that all of the fission products released from the damaged fuel 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.

3.1.6.2.1 Transport from the Containment release case

To evaluate containment releases for the RCCA ejection accident, the licensee assumed that all activity from the damaged fuel would release to and mix instantaneously and homogeneously in the containment volume. As specified in proposed TS 6.8.4.h, this activity was modeled to leak from the containment to the environment at an initial rate of 0.20 w/o per day for the first 24 hours, followed by a rate of 0.10 w/o per day for the remaining 29 days of the 30-day RCCA ejection accident analysis period. This assumption is consistent with Regulatory Position 6.2 of RG 1.183, Appendix H.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition using the methodology from SRP 6.5.2. This credit was applied to the radionuclides released using a removal coefficient of 0.1 per hour for aerosols and 5.58 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 release case

For the case in which the RCCA ejection accident results in secondary system activity releases, the licensee assumed that all activity from the breached and melted fuel is released 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 and MSSVs. The release of noble gases is assumed to occur without mitigation or reduction.

The licensee assumed a primary-to-secondary leak rate of 0.6 gpm total for all SGs with a maximum of 0.2 gpm to any one SG. This is in accordance with proposed change to the accident induced leakage performance criteria of the SG Program as described in proposed TS Section 6.8.4.j.b.2. The licensee has proposed that the criteria be changed from the current value of 1 gpm total through all SGs and 500 gpd (0.35 gpm) through any one SG, to a total of 0.6 gpm through all SGs and 0.2 gpm through any one SG. The operational limit on primary to secondary leakage limit in any one SG is 150 gpd at room temperature. Therefore, this proposed change continues to maintain approximately a 2-to-1 margin to the operational leakage limit specified in the TSs. In accordance with the regulatory guidance from RG 1.183, the licensee converted the volumetric leak rate to mass leak rates using a density of 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

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 SGs should be assumed to continue until shutdown cooling is in operation and releases from the SGs have been terminated." In accordance with RG 1.183, the licensee assumed that the primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F. The licensee determined that this would occur at 125.4 hours after the initiating event. The licensee assumed that radioactivity is released to the atmosphere via steaming from the SG ADVs and MSSVs until the RHR system is capable of removing decay heat. The licensee has determined that this would occur at 63 hours after the initiating event.

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 a portion of the postulated leakage is assumed to flash to vapor based on the thermodynamic conditions in the RCS and secondary coolant system immediately following plant trip when tube uncovery is postulated. The licensee assumed that the portion of leakage that immediately flashes to vapor rises through the SG bulk water into the steam space and is released without credit for any reduction due to scrubbing within the SG bulk water. The licensee assumed that the primary-to-secondary leakage would mix with the SG bulk water without flashing during periods of total tube submergence.

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 SGs is limited by the moisture carryover from the SGs." Accordingly, the licensee assumed that the radioactivity in the bulk water of the unaffected SGs 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 SGs for up to 30 minutes following a reactor trip. 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 calculated a flashing fraction of 11% 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 assumed that releases from the SGs would occur from the MSSV or ADV with the most limiting atmospheric dispersion factors.

3.1.6.3 CR Ventilation Assumptions for the RCCA Ejection Accident

The CR ventilation system as described in Section 9.9 of the Turkey Point, Units 3 and 4, UFSAR, provides assurance of CR habitability during normal operating conditions, anticipated operational occurrences, and design basis accident conditions. For the RCCA ejection accident analysis, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

For the secondary side release case, the CR will be isolated on a high radiation signal from the normal CR intake monitors. The licensee applied a 60-second delay to account for the time to reach the setpoint, signal processing and damper closure time. For the containment release case, the CR will be isolated on a high radiation signal from the containment monitors. The licensee applied a 60-second delay to account for the time to reach the setpoint, signal processing and damper closure time. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

The secondary release scenario credits CR isolation from a high radiation signal on the CR intake monitor. The TS setpoint for this instrument is 2 mR/hr. For additional conservatism, the licensee used an analytical setpoint of 5 mR/hr to account for measurement and test uncertainties. For the design basis fuel failure and core melt fractions, the licensee determined that the exposure rate at the intake monitor exceeded the analytical setpoint by approximately 35%. The licensee considered an additional scenario with fuel failure fractions less than the design values in which the analytical setpoint would not be reached. For this scenario the licensee assumed manual CR isolation after 30 minutes. The additional case showed that while the offsite dose consequences are lower, the longer time to establish CR isolation resulted in CR doses that, while slightly higher than the automatic isolation case, remain below the regulatory acceptance criteria.

3.1.6.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated RCCA ejection accident at Turkey Point Units 3 and 4, 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 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 used in the analysis are presented in Table 10 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the NRC staff determined that the doses estimated by the licensee for the Turkey Point Units 3 and 4, RCCA ejection accident, assuming either automatic or manual CR isolation, will comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.1.7 Waste Gas Decay Tank (WGDT) Rupture

This event involves a major rupture of one of the WGDT as described in Section 14.2.3 of the Turkey Point UFSAR. For this analysis the licensee assumes that the ruptured WGDT contains an inventory equivalent to the equilibrium RCS noble gas activity from operation with 1% fuel defects. The analysis is performed assuming a major rupture of the WGDT, which instantaneously releases the entire contents of the tank to the environment. The licensee did not credit hold-up, dilution, or filtration in the Reactor Auxiliary Building.

Since RG 1.183 does not provide direct guidance relative to the WGDT rupture event the licensee used the guidance provided in BTP 11-5 of the SRP, with additional instruction available from RIS 2006-04 to evaluate the WGDT rupture accident. BTP 11-5, with clarifications from RIS 2006-04, establishes the AST dose limit at 100 mrem TEDE at the EAB for this event. Since the WGDT rupture accident is modeled as an instantaneous puff release, the limiting time interval will, by definition, be the first 2 hours. For consistency, the licensee evaluated the LPZ dose against the 100 mrem TEDE EAB limit. In addition, for the evaluation of the CR, the licensee applied a limit of 5 rem TEDE, which is consistent with all AST CR evaluations.

3.1.7.1 WGDT Rupture Source Term

To conservatively bound the source term for the WGDT rupture analysis, the licensee assumed that the entire RCS noble gas inventory resulting from extended full power operation with 1% defective fuel is transferred to the WGDT following a shutdown. The source term consists of only noble gases since particulates and iodines are removed by other processes prior to transfer to the WGDT. The licensee calculated this inventory to be equal to 84,274.8 curies DEX, which exceeds the TS LCO 3.7.9 limit of 70,000 curies.

3.1.7.2 WGDT Rupture Transport

The WGDT tank area is served by the auxiliary ventilation exhaust fans that are not required to be operable by the TSs. Therefore, the licensee did not credit transport through the ventilation exhaust fans. The licensee analyzed this event by assuming a ground level release from the building which houses the WGDT, with no credit for effluent monitoring, isolation or for a building wake factor. The staff finds this transport model to be conservative because use of the ventilation exhaust fans would result in an elevated release model and greater dispersion of gases.

3.1.7.3 CR Ventilation Assumptions for the WGDT Rupture

The CR intake and recirculation filters do not remove noble gas isotopes. Therefore, to maximize the radiological effect on the CR, the licensee assumed that the CR would not be isolated during the analysis period. This assumption is conservative since during the normal mode of operation the CR intake flow rate is maximized. In addition, the atmospheric dispersion factors are higher when using the normal CR intake structure that will maximize the dose consequence.

3.1.7.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated WGDT rupture accident at Turkey Point Units 3 and 4, and concluded that the radiological consequences at the EAB meet the dose acceptance guidance provided in BTP 11-5 of the SRP. 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 used in the analysis are presented in Table 11 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the NRC staff determined that the doses estimated by the licensee for the Turkey Point, Units 3 and 4, WGDT accident will comply with the requirements of 10 CFR 50.67 and the guidelines provided in BTP 11-5 of the SRP, and are therefore acceptable.

3.1.8 Spent Fuel Cask Drop (SFCD) Accident

Postulated cask drop accidents at Turkey Point, Units 3 and 4, are described in UFSAR Section 14.2.1.3. that states, "The spent fuel transfer cask will not be moved into the spent fuel pit containing two region density racks until all spent fuel in the pit has decayed for a minimum of 1525 hours." The licensee evaluated this accident by assuming that all 157 assemblies of a recently discharged core are damaged by the cask drop.

The cask is assumed to impact the stored spent fuel assemblies and result in the release of fission products contained within the fuel gap of the stored fuel. No damage is postulated for the fuel being transferred in the transfer cask. The SFCD is not addressed in RG 1.183 or in SRP Section 15.0.1. The licensee used the methodology described in RG 1.183, Appendix B, which outlines the requirements for performing a radiological analysis of a FHA, to perform the dose consequence analyses for the SFCD accident. The licensee also applied the dose acceptance criteria for a FHA as described in SRP Section 15.0.1, Table 1 and RG 1.183, Table 6. It should be noted that in performing the radiological consequences of a SFCD in the SFP, the licensee did not take credit for the use of an impact limiting pad in the SFP.

SRP Section 15.7.4, "Radiological Consequences of a Fuel Handling Accident," Revision 1, July 1981, covers the review of the radiological consequences of a postulated fuel handling accident and states that, "Such accidents include the dropping of a single fuel assembly and handling tool or of a heavy object onto other spent fuel assemblies." Since the Turkey Point SFCD accident does not postulate the damage of the fuel being transferred in the transfer cask, the accident should be evaluated as the drop of a heavy object onto other spent fuel assemblies.

3.1.8.1 SFCD Source Term

For the SFCD accident, the licensee has defined the event as a cask drop onto stored spent fuel that results in the damage to all of the fuel pins in 157 fuel assemblies. The licensee evaluated the assembly source term for the SFCD using the same conservative approach used for the FHA with the exception of the decay times considered. The licensee's evaluation of the source term for the SFCD maximizes the activity for each isotope in each of the 157 damaged assemblies, which is conservative.

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis SFCD. 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.

The licensee modified the gap fractions specified in Table 3 of the RG 1.183 to account for high burnup fuel using the guidance from NUREG/CR-5009. The gap fractions from NUREG/CR-5009 as used by the licensee are approximately twice those specified in RG 1.183 and are conservative for use in the SFCD analysis.

3.1.8.2 SFCD Transport

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the SFP depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3 the licensee assumed that; the chemical form of radioiodine released from the fuel consists of 95% CsI, 4.85% I_2, and 0.15% organic iodine, the CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This results in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

Pursuant to guidance provided in RG 1.183, the licensee assumed that all of the fission products released from the SFP are released to the environment over a 2-hour period. The licensee modeled the release to the environment as a ground-level release. The licensee evaluated the SFCD based on the most limiting release location relative to the CR that was determined to be a release from the Unit 4 SFP. The licensee assumed no credit for filtration of the activity released from the SFP water prior to being released to the environment.

When the correction to the elemental iodine decontamination factor as discussed in Item 8 of RIS 2006-04 (ADAMS Accession No. ML053460347) is applied, RG 1.183, Appendix B, Regulatory Position 2, will be revised to 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). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 70% elemental and 30% organic species.

As noted previously, the licensee assumed a minimum water depth of 23 ft covers the underlying damaged fuel assembly in both the reactor cavity and SFP for the FHA analyzed in the subject LAR. The damaged fuel rods release 100% of the gap activity to the surrounding water. Consistent with the guidance in RG 1.183 for the FHA, the SFP or reactor cavity water cover provides a decontamination of the released iodine activity with an overall DF of 200. This DF results in 0.5% (i.e., 99.5% of the iodine is 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 be released per Regulatory Position 3 of RG 1.183.

3.1.8.3 CR Ventilation Assumptions for the SFCD

For the SFCD, the CR ventilation system is initially assumed to be operating in normal mode. The air flow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air make-up and an unfiltered inleakage of 100 cfm. The CR is assumed to be manually isolated by operator action 30 minutes after the initiating event. After CR isolation, the air flow distribution is assumed to consist of 525 cfm of filtered makeup flow from the more limiting of the two emergency outside air intakes, 100 cfm of assumed unfiltered inleakage, and 375 cfm of filtered recirculation flow. The licensee assumed a CR ventilation filter efficiency of 99% for particulates and 95% for elemental and organic iodine for both the filtered makeup and the recirculation flow.

3.1.8.4 Conclusion

The licensee evaluated the radiological consequences resulting from a postulated SFCD at Turkey Point, Units 3 and 4, 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 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 used in the analysis are presented in Table 12 and the licensee's calculated dose results are given in Table 1. Based on a review of the assumptions and methods discussed above, the NRC staff determined that the doses estimated by the licensee for the Turkey Point Units 3 and 4, SFCD will comply with the requirements of 10 CFR 50.67 and the guidelines of RG 1.183, and are therefore acceptable.

3.2 Atmospheric Dispersion Estimates

The licensee generated new CR, EAB, and LPZ atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of the limiting DBAs. Initially, the CR χ/Q values were based on Turkey Point onsite meteorological measurements made from 2003 through 2007. The EAB and LPZ χ/Q values were calculated using the data from 2005 through 2007. The licensee transmitted the data to the NRC with a description of the methodologies, other inputs, and assumptions used to calculate the χ/Q values. NRC review of the information provided by the licensee resulted in questions regarding data quality, processing, and application. While other areas of the analyses were also addressed, the primary area of focus related to the measurement of temperature difference with height, ΔT, which is used to estimate atmospheric stability. The licensee subsequently performed additional detailed assessments and provided several revisions to the data sets, analyses, and the CR, EAB and LPZ χ/Q values. An intermediate set of χ/Q values were generated based upon input files that deleted the meteorological data from 2003 and 2004, amended the data from 2005 through 2007, and appended data from 2008 and 2009. Subsequently, the licensee applied bias factors to the 2005 through 2009 ΔT measurements to reconcile use of instruments which were not consistently within the RG 1.23, "Onsite Meteorological Programs," Rev. 0, ΔT specification guideline and recalculated the χ/Q values. The resulting limiting χ/Q values identified by the licensee represent a change from those currently presented in Chapter 14 of the Turkey Point Unit 3, and Unit 4, UFSAR.

3.2.1 Meteorological Data

The licensee initially provided meteorological data by letter dated July 21, 2009. This information was provided in the form of hourly data from 2003 through 2007 that were formatted for input into the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). In addition, the licensee provided the meteorological data for the 2005 through 2007 period in the form of a joint wind speed, wind direction and atmospheric stability frequency distribution (JFD) for input to the PAVAN atmospheric dispersion computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radiological Materials from Nuclear Power Stations"). The data was measured primarily on the South Dade 60 meter meteorological tower, but, in some cases, measurements from the Land Utilization 10 meter meteorological tower were used as backup data. Wind direction, wind speed, and temperature were measured at heights of approximately 10 and 60 meters on the South Dade tower and at approximately 10 meters on the Land Utilization tower.

The NRC staff performed a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review of the hourly data and the JFD was performed using computer spreadsheets. The review identified several apparent inconsistencies, anomalies, and other areas of potential concern, particularly related to the ΔT measurements. As a result, the NRC staff issued a series of requests for information (RAIs) regarding the quality of the data with respect to measurement, processing, and the licensee's determination that the data were of high quality.

In response to the NRC RAIs, the licensee performed a reanalysis of the data and decided to revise the meteorological data set to omit years 2003 and 2004, amend the data in years 2005 through 2007, and append years 2008 and 2009, which were not available when the LAR was being prepared. By letter dated June 11, 2010, the licensee transmitted revised meteorological data for the 5-year period 2005 through 2009. In addition, the licensee initiated several corrective actions related to the current meteorological measurement program and acknowledged that further improvements in equipment, procedures, programs, and processes would be required to achieve performance objectives that are recommended in RG 1.23.

Because ΔT measurements from 2005 through 2009 were not consistently within the instrument specifications given in RG 1.23, the licensee performed a custom assessment of the ΔT measurements using calibration data. The assessment compared the reading of each measurement device (thermistor) with the temperature standard at the beginning and end of each 6-month calibration period that provided a basis to normalize the measurements. The licensee then decided to bias the ΔT measurements using the average (mean) calculated value for each channel over each calibration period of the hourly measurements. Each thermistor, which was replaced at each calibration, experienced variable drift during each period of measurement. The licensee assumed the drift to be linear when developing the bias factors. In response to questions from the NRC staff regarding basing the bias factors on the mean for each thermistor for each calibration period and the licensee's assumption that the drift was linear, the licensee proposed the use of the more limiting χ/Q values when the resultant biased χ/Q values exceeded the unbiased χ/Q values. In addition, the licensee stated that the manufacturer had provided information that the thermistors were expected to be linear in their behavior over the expected range of ambient air temperatures. The licensee provided results of a sensitivity study discussing use of several sets of bias factors that showed the impact on the dose was small with respect to this specific LAR. On September 2, 2010, the licensee submitted the revised meteorological data for 2005 through 2009 with the applied ΔT bias factors.

With regard to the NRC staff review of the 2005 through 2009 meteorological data that applied the ΔT bias factors, examination of the adjusted atmospheric stability data revealed that stable and neutral atmospheric conditions were usually reported to occur at night and unstable and neutral conditions during the day, as expected. The NRC staff continued to note a reported higher occurrence of extremely unstable conditions, stability class A, in 2005 and 2006, than in 2007 through 2009. Wind speed distributions were similar from year to year at each level, with the 60 meter wind speeds being faster than the 10 meter winds most of the time, as expected. Wind direction frequency occurrence was reasonably similar from year to year at both levels and between the two levels. The combined data recovery of the 10 meter wind speed, 10 meter wind direction, and stability data was in the mid- to upper-90 percentiles throughout the 5-year period, which was facilitated, to some extent, by use of back-up data. Wind direction measurement recovery at the 60 meter level did not meet the 90 percent recovery goal cited in RG 1.23, in part, because back-up wind direction and wind speed measurements are not made at the 60 meter level.

In summary, the NRC staff has reviewed the available information relative to the onsite meteorological measurements program, the 2005 through 2009 unbiased and biased meteorological data measured at the Turkey Point site, and the ARCON96 and PAVAN meteorological data input files provided by the licensee. On the basis of this review, the NRC staff has concluded that the data provides an acceptable basis for making estimates of atmospheric dispersion for the proposed AST DBA assessments associated with the current specific LAR. However, given that the licensee has initiated upgrades in the Turkey Point meteorological measurements program to ensure that measurements are of high quality, the NRC staff notes that the 2005 through 2009 data should not be considered acceptable for use in other licensing actions without further NRC staff review to ensure that the data are acceptable in the specific application for which they will be used.

3.2.2 CR Atmospheric Dispersion Factors

To assess the CR postaccident atmospheric dispersion conditions, the licensee generated χ/Q values using the ARCON96 computer code and guidance provided in RG 1.194. RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in design basis accident radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, release-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for Turkey Point.

The Turkey Point nuclear plant has one normal and two emergency CR air intakes. As a result, the licensee modeled more than 100 individual cases representing Unit 3 and Unit 4 release-receptor pairs, including those resulting from LOOP and other single failures. All sources were modeled as ground level point releases. The licensee calculated the straight-line horizontal distance between each release-receptor pair using the site location geodetic coordinates. The licensee stated that conservative release-receptor pairs were selected on a case-by-case basis in order to bound other possible release paths which may be applicable to each DBA.

The licensee initially used meteorological data measured from 2003 through 2007. Due to findings resulting from a detailed reassessment of this data, the licensee first recalculated the CR χ/Q values using amended data from 2005 and 2009. Subsequently, the licensee recalculated the CR χ/Q values using the 2005 through 2009 data base with biased ΔT data. Other assumptions and inputs were not changed.

Additional details on the licensee's assessments of control room postaccident atmospheric dispersion conditions are as follows:

- The licensee will relocate the two outside CR emergency air intakes as described in Section 3.4, Commitments. When in operation, the intakes will have balanced intake flow rates, capable of drawing outside makeup air from both locations. The two intakes will be adequately separated with respect to criteria in RG 1.194 to permit reduction of the χ/Q values for the more conservative intake by a factor of two in the accident analyses for all release points in separate wind sectors.

- To model the χ/Q value for the limiting ADV or MSSV release prior to the beginning of reactor coolant system cool down, the licensee assumed plume rise and reduced the ground level χ/Q values calculated using ARCON96 by a factor of five. RG 1.194 states that this reduction may be taken only if the release point is uncapped and vertically oriented and the time-dependent vertical velocity exceeds the 95th-percentile wind speed at the release point height by a factor of five.

- The licensee stated that the effluent releases from the MSSV and ADV silencers are uncapped and oriented in a vertical upward direction. The licensee evaluated and extrapolated the 10 meter wind speeds to estimate a 95th percentile wind speed of 16.8 miles per hour (24.6 feet per second (fps)) at the 18.6 meter release height for the limiting release location. The NRC staff assessed the 2005 through 2009 meteorological data and concluded that the licensee's estimate is reasonable. Thus, to ensure a ratio of at least a factor of five, the minimum vertical effluent exit speed at any time from any MSSV or ADV would need to be at least 123 fps. The licensee stated that MSSV and ADV vertical exit velocities, which are determined from hot zero power plant conditions and are based upon the rated relief valve capacities, are greater than 125.7 fps and 194.0 fps, respectively. The NRC staff approximations confirm that the ratio of the stated minimum effluent exit speed to the 95th percentile wind speeds is greater than a factor of five.

- The horizontal straight line distance between the Unit 4 steam jet-air ejector (SJEA) and the normal CR intake is 9.4 meters. RG 1.194 states that ARCON96 should not be used for release-receptor distances less than about 10 meters and advises that such a situation should be addressed on a case-by-case basis. As a result, the licensee generated an alternative χ/Q value for the SJEA-normal intake pair by calculating χ/Q values for hypothetical receptors at distances of 10 and 20 meters using ARCON96 and applying adjustment factors based upon the squares of the distances. This resulted in a value approximately 15 percent higher than that using ARCON96 for an input horizontal distance of 9.4 meters. The NRC staff finds the resultant χ/Q value approximation acceptable in this specific case given that the shortest distance between the SJEA and normal intake is only slightly less than 10 meters when the differences in height are also factored in.

The NRC staff has reviewed the licensee's assessments of control room postaccident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. This included a review of the inputs and assumptions, which the NRC staff found generally consistent with site configuration drawings and input tables, and the NRC staff practice. In addition, the NRC staff generated sample comparative χ/Q value estimates and found the resultant χ/Q values to be similar to those calculated by the licensee for the cases considered.

On the basis of this review, the NRC staff has concluded that the χ/Q values in Table 2 are acceptable for use in DBA control room dose assessments addressed in this LAR. However, the NRC staff notes that any future calculations of CR χ/Q values should consider use of meteorological data from the upgraded Turkey Point meteorological measurements program or a reassessment of the 2005 through 2009 data set to ensure that the data is acceptable in the specific application for which they will be used.

3.2.3 Offsite Atmospheric Dispersion Factors

The licensee calculated EAB and LPZ χ/Q values using guidance provided in RG 1.145 and the PAVAN atmospheric dispersion 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). 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. The licensee assumed a minimum containment cross-sectional area of 1254 square meters and a containment height of 46.4 meters above ground level. Other inputs included the Turkey Point EAB distance, which is variable as a function of direction and the LPZ distance which is 8045 meters in all directions.

For each time interval, the licensee compared the maximum χ/Q value from all downwind sectors with the 5 percentile over site χ/Q value and selected the higher of the two as the limiting value at the EAB and LPZ during each respective time interval. However, with the exception of the waste gas decay tank rupture, only the 0-2 hour EAB χ/Q value was used in the DBA analyses for the calculation of the EAB dose.

The NRC staff qualitatively reviewed the inputs and assumptions used in the licensee's PAVAN computer calculations and generated comparative χ/Q values. The review resulted in questions to the licensee regarding the format and quality of the meteorological data. As a consequence, the licensee provided two successive revisions to the EAB and LPZ χ/Q values as follows.

- The licensee initially used meteorological data measured from 2005 through 2007, which was formatted into a JFD using seven wind speed categories which were generally consistent with the example provided in RG 1.23, Rev. 0. However, this choice of wind speed categories appeared to result in some clustering of the data in the lower categories which could affect the resultant χ/Q value estimates generated by the PAVAN computer code. RIS 2006-04 recommends that input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results. Therefore, the licensee developed a revised JFD, based upon the 2005 through 2007 data, which redistributed the data into 14 wind speed categories consistent with the RIS recommendations and generated revised EAB and LPZ χ/Q values.

- Due to subsequent findings resulting from the reassessment of the 2003 through 2007 data, the licensee provided a second revision to the EAB and LPZ χ/Q estimates using the amended 2005 through 2009 meteorological data files. Other assumptions and inputs were not changed when the licensee recalculated the CR χ/Q values. The NRC staff generated a JFD from the 2005 through 2009 data which applied the biased ΔT data which the licensee had provided to the NRC in the ARCON96 hourly format. The NRC staff then calculated comparative χ/Q values and found their results to be similar to the χ/Q values calculated by the licensee.

Therefore, on the basis of this review, the NRC staff has concluded that the resulting EAB and LPZ values generated by the licensee and presented in Table 3 of this SE are acceptable in the DBA dose assessments performed in support of this LAR. These X/Q values represent a change from those used in the current licensing basis. However, the NRC staff notes that any future calculations of EAB or LPZ χ/Q values should consider use of meteorological data from the upgraded Turkey Point meteorological measurements program or a reassessment of the 2005 through 2009 data set to ensure that the data are acceptable in the specific application for which they will be used.

3.3 Structural Integrity

The NRC staff's review of the licensee's application for the full scope AST implementation focused primarily on the structural integrity of SSCs that are credited in the proposed AST. Specifically, the structural integrity of portions of the CREVS, which was credited in the licensee's application for an AST, were evaluated to determine whether the proposed modifications to the system would maintain continued safe operation under the design basis loading conditions, following the proposed AST implementation. Additionally, in accordance with the Turkey Point design basis requirements, the NRC staff reviewed the licensee's proposed addition of 10 stainless steel baskets, containing NaTB, to determine the acceptability of the design of the new baskets and to ensure that the potential for seismic interaction of the new baskets with Seismic Class I equipment presently inside containment has been addressed. The NRC staff also reviewed the potential effects on the structural integrity of the NCC coils, due to the proposed modifications to the NCC coils, to evaluate the acceptability of the proposed modifications.

3.3.1 Control Room Emergency Ventilation System Structural Evaluation

3.3.1.1 Control Room Emergency Ventilation System Duct Relocation

By letter dated June 25, 2009, Section 3.0 of Enclosure 1, the licensee described the proposed changes that will be implemented in conjunction with the proposed AST implementation at Turkey Point. Included in these changes is the proposed relocation of the CREVS intakes to new locations for the purposes of reducing the impact of postaccident atmospheric contaminants by diversifying the CREVS intake locations and creating a dual intake system. The licensee has committed to relocating the aforementioned CREVS intakes prior to the AST implementation, as indicated in Section 5.0 of Enclosure 1 to Reference 1.

Section 9.9.1 of the Turkey Point UFSAR describes the CREVS design basis and operational characteristics. By letter dated August 25, 2009, the licensee indicated that the modified CREVS intake components would be classified as Seismic Class I components, consistent with the existing classification for the CREVS. Accordingly, the licensee indicated in the same response that the CREVS modifications would be designed in accordance with the structural design criteria and applicable codes and standards for Class I SSCs, which are identified in Appendix 5A of the Turkey Point UFSAR. Commensurate with the classification of the modified CREVS intake components as Seismic Class I, the loading combinations for which the licensee must consider when evaluating the structural adequacy of these components include SSE loads, ensuring that these components will remain functional following a design basis earthquake.

In response to an NRC staff RAI regarding the design and location of the relocated CREVS intakes, the licensee confirmed in its February 10, 2010, letter, that portions of the relocated CREVS intakes would be external and above grade in order for the proposed design to meet the atmospheric dispersion design requirements necessary for the AST implementation. The licensee indicated in its August 25, 2009, letter, that external, above grade portions of the relocated CREVS intakes would also be designed against applicable external missile criteria for Class I SSCs (Appendix 5E of Turkey Point UFSAR).

The maximum hypothetical accident (MHA), related to the external missile protection requirements for the external CREVS components, result from a postulated tornado-generated missile, as indicated in Appendix 5E of the Turkey Point UFSAR. The tornado-generated missile MHA design requirements found in this appendix stipulate that SSCs, such as the CREVS, which may be subject to an MHA missile impact, must be able to perform their designated function following any postulated impact. Specifically, the design must consider the postulated impact of a corrugated sheet of siding, wood decking, and a passenger car on the ground, each having a specified weight and impact velocity. The licensee confirmed in its February 10, 2009, letter, that these postulated external missiles were considered in the CREVS intake design.

The results of the licensee's structural evaluation of the relocated CREVS intake components were summarized in the February 10, 2010, letter. The licensee indicated that the CREVS intake components that were not subject to external missile impacts were evaluated against the applicable design requirements for Seismic Class I SSCs. The evaluation results demonstrated that, when subjected to the applicable loading combinations described in Appendix 5A of the Turkey Point UFSAR, the CREVS components meet the applicable acceptance criteria and design requirements. Additionally, for those portions of the CREVS intakes that are subject to postulated missile impact evaluations, the licensee indicated that these evaluations demonstrate that these CREVS intake components meet the applicable acceptance criteria. It was noted in the February 10, 2010, letter, that additional protection, in the form of bollards, barriers, or enclosures, would be provided for external CREVS components subject to the postulated passenger car missile. With regards to the additional design acceptance criteria for Class I SSCs, the licensee stated that the postulated missile impact evaluations were considered bounding based on the fact that the missile impact loadings are more severe than the loading combinations identified in Turkey Point UFSAR Appendix 5A for Class I SSCs.

3.3.1.2 Control Room Emergency Ventilation System Compensatory Filtration Unit

In its May 21, 2010, supplemental letter, the licensee indicated that a modification to the CREVS system would be undertaken that would involve the installation of a compensatory filtration unit. This unit is being installed to address an NRC staff concern regarding the potential inability of the CREVS to mitigate the consequences of a DBA if a currently-installed CREVS filter train becomes inoperable. In its May 21, 2010, letter, the licensee indicated that the compensatory unit would be designed as a safety-related, Seismic Class I system acting as a backup (i.e., compensatory) to the CREVS. In response to an NRC staff RAI that requested additional details regarding the structural design and seismic qualification of the compensatory filtration unit, the licensee confirmed in its September 15, 2010, letter, that the compensatory filtration unit would be designed as a Seismic Class I SSC in accordance with the Turkey Point licensing basis for these SSCs found in Appendix 5A of the Turkey Point UFSAR. As such, the unit will be designed to withstand the loads induced by a design basis earthquake and maintain its structural adequacy following this event.

In its RAI, the NRC staff also requested the licensee to provide the results of the seismic qualification performed for the compensatory filtration unit. However, based on the fact that the necessity of the compensatory filtration unit was determined after the submittal of the original AST LAR, the licensee indicated that the design details of the unit were not readily available and, therefore, a full structural analysis, including seismic qualification, of the unit had not yet been performed. Given that the NRC staff is relying on the seismic qualification of SSCs credited in the proposed AST methodology to assess whether the LAR has met the pertinent regulatory requirements, a License Condition will be required to ensure that these regulatory requirements are satisfied prior to the implementation of the proposed AST methodology. Therefore, the following License Condition will be required:

> The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey Point, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A- 1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

3.3.1.3 Conclusion

Based on the evaluation described above, which indicates that the relocated CREVS intake components meet the applicable structural design criteria for Seismic Class I SSCs, including those CREVS components subject to postulated MHA missile impacts, as described in the Turkey Point UFSAR, the NRC staff considers the structural aspects of the relocated CREVS intake design acceptable as they relate to the proposed AST implementation. The NRC staff will establish a License Condition to require that the structural design of the compensatory filtration unit satisfies the criteria for Seismic Class I SSCs to ensure that the regulatory requirements pertinent to this equipment are met. Given these considerations, the NRC staff is satisfied that reasonable assurance of the structural integrity of these SSCs will be established prior to the implementation of the proposed AST at Turkey Point.

3.3.2 Sodium Tetraborate Decahydrate Basket Structural Evaluation

By letter dated June 25, 2009, Section 4.2.10 of Enclosure 1, the licensee described proposed TS 3/4.6.2.3, regarding post-LOCA containment sump pH control measures utilizing NaTB, which will be implemented in conjunction with the proposed AST at Turkey Point. Appendix A to RG 1.183 requires licensees proposing to implement an AST methodology to maintain the post-LOCA recirculation sump water at a pH level of 7.0 or greater in order to minimize stress corrosion cracking of austenitic stainless steel components which may be exposed to the recirculation sump environment and to prevent re-evolution of radioactive iodine found in the recirculation sump water. To achieve this pH level, the licensee has proposed to install 10 (2 large, 8 small) passive, stainless steel baskets in the lower region of the Turkey Point containment structures. These baskets would hold a TS-minimum required amount (combined weight) of NaTB, which would be distributed passively to the sump water as the sump level rises following a LOCA. The licensee has committed to installing these baskets prior to the AST implementation, as indicated in the June 25, 2009, letter, Attachment 4 of Enclosure 1.

In response to an NRC staff request for supplemental information regarding the design details of the baskets to support the acceptance review of the proposed LAR, the licensee indicated in its August 26, 2009, letter, that the baskets would be designed to withstand Seismic Class III loads and load combinations in addition to avoiding interaction with Seismic Class I SSCs within the lower containment region where the baskets will be located. Additionally, the licensee stated that the baskets would be located away from postulated High Energy Line Break (HELB) regions of influence such that any potential HELB dynamic effects loadings do not affect the baskets. The licensee also stated that the baskets design will enable the structures to resist hydraulic flow-induced motion. In response to the NRC staff's RAI concerning the construction and structural evaluation of the baskets, the licensee indicated in its February 10, 2010, letter, that the baskets would be free-standing, steel-framed, mesh structures. While the baskets are free-standing structures, the design of the baskets stipulates that they will rest on four, stainless steel leveling casters, which can be locked in place. Based on the passive nature of the baskets and their general construction, the specified function of the baskets is not precluded in the event that an SSE level loading event renders a portion of the basket structurally damaged due to the fact that the NaTB will still be absorbed into the post-LOCA recirculation sump water. Therefore, the NRC staff considers the licensee's subjection of the baskets to Class III loads in their structural evaluation acceptable and consistent with Appendix 5A of the Turkey Point UFSAR, as it defines the classification of SSCs at Turkey Point.

In its August 26, 2009, letter, the licensee indicated that the design codes of record to be used for the NaTB baskets would include both the American Society of Mechanical Engineers Code and the American Institute of Steel Construction Manual for Steel Construction. Moreover, in the licensee's February 10, 2010, letter, it was stated that the maximum joint deflections, member stresses, bolted connections, welded connections, and basket mesh were compared against the code allowable values for the loads and load combinations identified above and found to be acceptable.

In addition to the Class III loads and load combinations to which the licensee structurally evaluated the baskets for, the design requirements outlined in Appendix 5A of the Turkey Point UFSAR indicate that Class III SSCs must be evaluated for earthquake loads if the potential for interaction with Safety Related SSCs exists. In the licensees RAI response found in its February 10, 2010, letter, the licensee presented the results of their sliding and overturning analyses demonstrating the ability of the baskets to resist sliding and overturning, such that any possible SSC interaction is precluded. In these analyses, the licensee demonstrated that, for the two large baskets and eight small baskets, the friction force resulting from the dead weight of the baskets (empty and full) had a significant amount of margin when compared to the sliding force induced by SSE loadings. The licensee also demonstrated that the resisting moments inherent in the basket weights are acceptable as compared to the induced overturning moments (performed only for the full baskets, which bounds the empty baskets).

Based on the evaluation described above, which indicates that the baskets meet the applicable structural design criteria, as described in the Turkey Point UFSAR, and the licensee's analyses results showing that the basket design precludes the possibility of basket interaction with Safety Related (and other) SSCs during seismically induced loadings due to an SSE event, the NRC staff determined that the structural aspects of the basket design are acceptable as they relate to the licensee's use of these baskets for containment sump pH control.

3.3.3 Normal Containment Cooler Coil Structural Evaluation

As described Section 3.3.2 above, the licensee has proposed to install 10 passive, NaTB-containing, stainless steel baskets in the lower region of the Turkey Point containment structures to achieve an increased, post-LOCA sump pH level. In the licensee's application dated June 25, 2009, Section 4.2.10, the licensee also detailed its proposed modifications to the NCC coils to offset the increased chemical debris generation due to the higher pH levels present in the post-LOCA sump environment. The proposed modifications to the NCC coils include the replacement of the current aluminum fins on the coils with copper fins. The licensee indicated that the current aluminum fins found on the NCC coils are the primary source of chemical debris in containment and the replacement of these fins with a copper material will generate less chemical debris. Additionally, this modification will maintain debris generation to levels within the design basis allowable values for the Turkey Point sump strainers. The licensee has committed to replacing the current aluminum fins prior to the AST implementation, as indicated in its June 25, 2009, application.

In response to an NRC staff request for supplemental information regarding the potential structural effects of the proposed NCC coil fin replacement, the licensee indicated in Reference 2 that the NCC coils are classified as Seismic Class I components and, as such, must be designed in accordance with the structural design criteria and applicable codes and standards for Class I SSCs. The applicable codes and standards for Class I SSCs are identified in Appendix 5A of the Turkey Point UFSAR. Also, in its August 26, 2010, letter, the licensee indicated that the fins are not pressure boundary components and that replacement fins have comparable or improved mechanical properties compared with the current fins (i.e., the yield and ultimate strengths of copper are generally higher than that of aluminum). In response to an NRC staff RAI similar to the request for supplemental information, the licensee confirmed in its letter dated February 10, 2010, that no credit would be taken for the structural capacity of the fins. The licensee also indicated that copper fins would be utilized in four replacement NCC units (including NCC coils and associated fins) to be installed in support of a proposed EPU at Turkey Point. However, the licensee indicated that this modification is not required for the proposed AST implementation. Based on the licensee's assertion that larger NCC units are not necessary for AST implementation, the NRC staff did not review the licensee's proposed modification to install larger NCC units in support of their proposed EPU and therefore, this license amendment does not constitute acceptability of these larger NCC units and the possible structural effects due to their installation. Based on the fact that the mechanical properties of the NCC coil fin material will be comparable or improved compared to the current fin material, the NRC staff considers the licensee's use of copper fins in lieu of aluminum fins acceptable for the purposes of AST implementation.

3.3.4 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed LAR associated with the implementation of the full scope AST methodology at Turkey Point on portions of the CREVS, the new NaTB baskets, and the NCC coils. As indicated above, the NRC staff will establish a License Condition that requires the design of the compensatory filtration unit, to satisfy the stress limits provided in Appendix 5A of the Turkey Point UFSAR for Seismic Class I SSCs. The removal of the licensing condition related to the structural design and seismic qualification of the compensatory filtration unit is contingent upon the design of the unit satisfying these stress limits. On the basis of the NRC staff's review as described above, which demonstrates the structural adequacy of the proposed modifications supporting the AST implementation, the NRC staff finds the proposed AST implementation acceptable. This acceptance is based on the demonstration that the proposed modifications, supporting the AST implementation, meet the intent of the aforementioned regulatory requirements related to the civil and mechanical engineering purview, which provides reasonable assurance that these SSCs will be able to perform their intended functions under their associated design-basis loading conditions following the implementation of the AST methodology.

3.4 Electrical Systems

The NRC staff has reviewed the electrical and electrical equipment environmental qualification portions of the license amendment request, and determined that nonsafety related electrical systems were not credited in the AST analyses.

The NRC staff requested additional information on whether any loads were being added to the Turkey Point emergency diesel generators (EDGs) and if so, how the loads being added would affect the capability and capacity of the EDGs as well as provide any changes to the loading sequence. The licensee stated in its March 15, 2010, letter, that no loads were added to the Turkey Point EDGs as a result of the AST adoption. The LAR specifies a proposed modification to delete the current Emergency Containment Filtration (ECF) System, which reduces the EDG loading (sequenced) by about 110 kilowatts per EDG. No changes to the EDG surveillance testing are proposed. The staff finds this design acceptable.

The NRC staff requested the licensee to provide a list and descriptions of components added to its 10 CFR 50.49 program due to the AST and, additionally, confirm that these components are qualified for the environmental conditions expected postevent. In its March 15, 2010, letter, the licensee stated that no components were added to the Turkey Point 10 CFR 50.49 program as a result of the AST adoption. The NRC staff finds this acceptable.

The NRC staff requested additional information regarding the proposed utilization of NaTB to control pH at 7.0 or higher, crediting Containment Spray for post-LOCA iodine removal and not crediting the Containment Filtration System in the AST dose consequences analyses and the impact of the above proposed actions on environmental conditions and any impact on equipment qualification. The licensee responded in its March 15, 2010, letter, that the utilization of the new passive system to release NaTB to control post-LOCA sump pH will continue to assure that the appropriate sump pH is maintained in the post-LOCA environment during the sump recirculation phase consistent with and bounded by the current equipment qualification pH profiles. In addition, the substitution of Containment Spray for ECF for post-LOCA iodine removal function will not impact environmental conditions within containment or impact any existing environmental qualification requirements. The NRC staff finds this acceptable.

The NRC staff also reviewed the environmental qualification portion of the license amendment request. The licensee used the methodology contained in TID 14844 to determine the radiation doses in the existing environmental qualification analyses as stated in Enclosure 1, Section 4.2

of the licensee's June 25, 2009, letter. As mentioned previously, the use of this methodology is consistent with the guidance contained in RG 1.183. Since the licensee will continue to use the TID 14844 methodology and no new equipment is added to its 10 CFR 50.49 program, the NRC staff determined that the environmental qualification of equipment should remain bounding during full-scope implementation of an AST.

3.5 Technical Specification Changes

3.5.1 TS Definitions Section 1.12, Definition of Dose Equivalent Iodine (DEI)

The licensee has proposed to revise the definition of DEI in TS Section 1.12 to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," as the source of thyroid dose conversion factors.

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 committed dose equivalent (CDE) thyroid DCFs as opposed to the committed effective dose equivalent (CEDE) DCFs, both of which are listed in Table 2.1 of FGR 11.

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 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 Table 2.1 of FGR No.11 in the DEI definition is acceptable.

3.5.2 Deletion of the Definition of E Bar and the addition of a new definition for DE Xe-133

It should be noted that justification for the following TS changes related to the deletion of E Bar and the addition of a new definition for DE Xe-133 generally follow NRC staff's model safety evaluation for TS Task Force TSTF-490, Revision 0 "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec."

TS Section 1.13 definition for E - Average Disintegration Energy (E Bar) is deleted and replaced with a new definition for DOSE EQUIVALENT XE-133 (DEX) that states:

> "DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT

XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of Environmental Protection Agency FGR No. 12, 1993, 'External Exposure to Radionuclides in Air, Water, and Soil.'"

The new definition for DEX is similar to the definition for DEI. The determination of DEX will be performed in a similar manner to that currently used in determining DEI, except that the calculation of DEX is based on the acute dose to the whole body and considers the noble gases Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138, which are significant in terms of contribution to whole body dose. Some noble gas isotopes are not included due to low concentration, short half life, or small dose conversion factor. Using this approach, the limit on the amount of noble gas activity in the primary coolant would not fluctuate with variations in the calculated values of E Bar. If a specified noble gas nuclide is not detected, the new definition states that it should be assumed the nuclide is present at the minimum detectable activity. This will result in a conservative calculation of DEX.

When E Bar is determined using a design basis approach in which it is assumed that 1.0% of the power is being generated by fuel rods having cladding defects and it is also assumed that there is no removal of fission gases from the letdown flow, the value of E Bar is dominated by Xe-133. The other nuclides have relatively small contributions. However, during normal plant operation there are typically only a small amount of fuel clad defects and the radioactive nuclide inventory can become dominated by tritium and corrosion and/or activation products, resulting in the determination of a value of E Bar that is very different than would be calculated using the design basis approach. Because of this difference, the accident dose analyses become disconnected from plant operation and the LCO becomes essentially meaningless. It also results in a TS limit that can vary during operation as different values for E Bar are determined.

The current LCO 3.4.8 specifies the limit for primary coolant gross specific activity as 100/E Bar µCi/gm. The current E Bar definition includes radioisotopes that decay by the emission of both gamma and beta radiation. This change will implement an LCO that is consistent with the whole body radiological consequence analyses which are sensitive to the noble gas activity in the primary coolant but not to other non-gaseous activity currently captured in the E Bar definition.

The change incorporating the newly defined quantity DEX is acceptable to the NRC staff from a radiological dose perspective since it will result in an LCO that more closely relates the noniodine RCS activity limits to the dose consequence analyses that form their bases. The licensee has maintained consistency by using the same dose conversion factors in the formulation of the DEX LCO value as were used in the dose consequence analyses.

3.5.3 Modification to TS 3/4.4.8 RCS Specific Activity

The licensee has proposed to reduce the current RCS specific activity LCO for DEI-131 from "less than or equal to 1.0 micro curie per gram" to "less than or equal to 0.25 micro curies per gram." In addition, the licensee has proposed an LCO of "less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133," replacing the current LCO related to the deleted E Bar ($\bar{E}$).

The licensee's proposed limit for DEI-131 is more restrictive than the limit currently in place and consistent with that assumed in the proposed AST accident dose consequences analysis. The licensee states that the proposed TS limit for DEI-131 is approximately 40 times higher than the highest measured DEI-131 from previous operating cycles that contained fuel failures and that reducing the RCS specific activity limit from 1.0 microcurie per gram to 0.25 microcurie per gram of DEI-131 will not result in an undue burden on plant operation. Reducing the RCS specific activity limit for DEI places a more conservative operating condition on the licensee, which is acceptable to the NRC.

The licensee has proposed to change TS LCO 3.4.8.b by replacing the current limit of 100/Ē with DOSE EQUIVALENT XE-133 less than or equal to 447.7 microcuries per gram. This limit is established based on the RCS activity corresponding to 1% fuel clad defects with sufficient margin to accommodate the exclusion of those isotopes based on low concentration, short half life, or small dose conversion factors and is consistent with that assumed in the accident dose consequences analysis. The primary purpose of the TS 3.4.8 LCO on RCS specific activity and its associated actions is to support the dose analyses for DBAs. The whole body dose is primarily dependent on the noble gas activity, not the nongaseous activity currently captured in the Ē definition. The NRC staff has performed confirmatory calculations and agrees that the proposed limit of DOSE EQUIVALENT XE-133 of less than or equal to 447.7 microcuries per gram will accurately reflect the initial conditions of the design basis dose consequence analyses submitted for the AST LAR and is therefore acceptable.

The licensee has proposed to modify the Applicability of TS 3.4.8 from the current MODES 1, 2, 3, 4, and 5 to MODES 1, 2, 3, and 4. It is necessary for the LCO to apply during MODES 1 through 4 to limit the potential radiological consequences of an SGTR or MSLB that may occur during these MODES. In MODE 5 with the RCS loops filled, the SGs are specified as a backup means of decay heat removal via natural circulation. In this mode, however, due to the reduced temperature of the RCS, the probability of a DBA involving the release of significant quantities of RCS inventory is greatly reduced. Therefore, monitoring of RCS specific activity is not required. In MODE 5 with the RCS loops not filled, and in MODE 6, the SGs are not used for decay heat removal, the RCS and SGs are depressurized and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required. The change to modify TS 3.4.8 Applicability to exclude MODE 5 but continue to include MODE 1, 2, 3, and 4 will limit the potential radiological consequences of an SGTR or MSLB that may occur during these MODES and is, therefore, acceptable to the NRC staff from a radiological dose perspective.

The licensee has proposed to delete Figure 3.4.-1, "DEI-131 Reactor Coolant Specific Activity Limit vs. Rated Thermal Power with the Reactor Coolant Specific Activity > 1µci/gram DEI-131." The current full power limit of 60 microcuries per gram DEI-131 will be used at all power levels. The licensee has proposed to change Action a. to delete the reference to Figure 3.4.-1 and state that with the specific activity of the reactor coolant greater than 0.25 microcuries per gram DEI-131, verify DEI-131 is less than or equal to 60 microcuries per gram once per 4 hours. The change from a graph that allows the DEI-131 spiking limit to vary with power level to a specific value is consistent with dose consequence events that are analyzed at full-power and assume a pre-accident spike of 60 microcuries per gram DEI-131. The full power transients that allow a DEI-131 spike are not changed. The curve contained in Figure 3.4.-1 was provided in a June 12, 1974, letter from the AEC, "Proposed Standard Technical Specifications for Primary Coolant Activity." Radiological dose consequence analyses for SGTR and MSLB accidents that take into account the pre-accident iodine spike do not consider the elevated RCS iodine specific activities permitted by the existing Figure 3.4.-1 for operation at power levels below 80% RTP. Instead, the pre-accident iodine spike analyses assume a DEI-131 concentration corresponding to the short term site specific DEI spiking limit which the licensee has established as 60 µCi/gm DEI-131. The NRC staff agrees with the proposed change and that TS 3.4.8 Required

Action A.1 should be based on the short term site specific DEI-131 spiking limit to be consistent with the assumptions contained in the radiological consequence analyses.

The licensee has proposed new Actions a. through e. which are incorporated to implement the new LCOs for DEI and DEX and to replace Figure 3.4-1 with a specific value for DEI of 60 microcuries per gram. Actions a. through c. provide the requirements for DEI while Actions d. and e. provide the requirements for DEX. The licensee has maintained consistency with the required actions and completion times in the current TSs. The licensee has proposed to revise Actions c. and e. to include a requirement to be in COLD SHUTDOWN within the next 30 hours. This requirement will ensure the unit is placed in a Mode where the TS is not applicable.

In addition, the licensee has proposed to revise Actions b. and d. to state that LCO 3.0.4. is not applicable. This will allow entry into a Mode or other specified condition in the LCO Applicability when LCO 3.4.8 is not being met. The proposed change to Action b. would allow entry into the applicable Modes from MODE 4 (HOT SHUTDOWN) to MODE 1 (POWER OPERATION) while the DEI is greater than the 0.25 microcuries per gram and less than or equal to 60 microcuries per gram and DEI is being restored to within its limit. The change to Action d. would allow entry into the applicable Modes from MODE 4 (HOT SHUTDOWN) to MODE 1 (POWER OPERATION) while the DOSE EQUIVALENT XE-133 is greater than 447.7 microcuries per gram and the DEX is being restored to within its limit. The NRC staff finds that this Mode change is acceptable due to the significant conservatism incorporated into the DEI and DEX specific activity limits, the low probability of an event occurring that is limiting due to exceeding the specific activity limits, and the ability to restore transient specific excursions while the plant remains at, or proceeds to power operation.

The licensee has proposed to change TS Table 4.4-4, Item 5 from "Radiochemical for $\bar{E}$ Determination" to "Isotopic Analysis for DOSE EQUIVALENT XE-133." This surveillance requires a gamma isotopic analysis as a measure of the noble gas activity of the reactor coolant at least once every 7 days. The measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. The surveillance provides an indication of any increase in noble gas specific activity. Trending the results of this surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The surveillance frequency of 7 days considers the low probability of a gross fuel failure during this time. If a specific noble gas nuclide listed in the definition for DOSE EQUIVALENT XE-133 is not detected, it will be assumed to be present at the minimum detectable activity. The licensee also has proposed several administrative changes to accommodate replacing the $\bar{E}$ determination with DEX. The NRC staff finds that these changes are consistent with TSTF-490 for replacing the $\bar{E}$ determination with DEX and are therefore acceptable.

In a letter dated May 21, 2010, the licensee responded to a request for additional information to describe what methods, procedures, etc. are in place to ensure that LCO 3.4.8 will not be exceeded in all the applicable modes given that the surveillance requirement (SR) is stated as only being applicable in Mode 1. The licensee responded that although the TS SR 4.4.8 does not explicitly require the SR to be performed in MODES 2, 3, and. 4, the surveillance requirement is still required to be met during the Modes of Applicability (MODES 1, 2, 3, and 4) in accordance with LCO 3.0.1. The licensee stated that although DEX is not measured for thermal power changes, DEI is measured during thermal power changes greater than 15% and that if at any time during MODES 1 through 4 there is information or plant indication that LCO 3.4.8 may not be met, the SR would be performed to ensure that there is not a failure to meet the LCO.

The licensee also stated that after review of recent TSTF-490 submittals for North Anna, Three Mile Island, and Kewaunee and after additional discussion with the NRC regarding the surveillance requirement, the licensee agreed to revise the applicable modes in which sampling and analysis is required in TS Table 4.4-4 from MODE 1 to MODES 1, 2, 3, and 4 for Items 3 and 5 on DEI and DEX, respectively. The licensee asserts that this change will provide explicit requirements and clear direction to the operating staff. Therefore, with this change, the surveillance will be explicitly required to be performed during the Modes of Applicability (MODES 1, 2, 3, and 4), which will ensure the potential consequences of a steam line break or SGTR are bounded by the approved accident analyses.

3.5.4 TS 3/4.6.3 Emergency Containment Filtering System

The licensee has proposed to delete the ECF System TS since this system is not credited in the AST analyses. With the acceptance of this AST LAR this system is no longer required to mitigate any DBA. The licensee asserts that the ECF System iodine removal capability is no longer required since the dose consequences of the revised AST analyses that do not credit the ECF are within regulatory limits. As such, the ECF system does not meet the 10 CFR 50.36 criteria for inclusion in the TSs. As stated in 10 CFR 50.36, a TS LCO must be established for each item meeting one or more of the following criteria:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment [PRA] has shown to be significant to public health and safety.

Criterion 1 and Criterion 2 were never applicable to the ECF System. Criterion 3 is no longer applicable to the ECF since the AST analysis does not credit the ECF system. Criterion 4 is not applicable to the ECF System since the ECFs are not modeled in the licensee's PRA. Credit for the ECF System will be removed from the Turkey Point Unit 3 and 4 licensing basis. The proposed change in this LAR will delete TS 3/4.6.3, ECF System. Due to the above, the NRC staff finds this proposed change to delete TS 3/4.6.3, ECF System, to be acceptable from a dose consequence perspective.

3.5.5 TS 3/4.7.5 Control Room Emergency Ventilation System

The licensee has proposed a change to the TS requirements related to CRE habitability in TS 3/4.7.5, "Control Room Emergency Ventilation System (CREVS)." The NRC staff understands that these TS changes are being proposed as a result of AST assumptions crediting the CREVS, and to address mitigating actions that will be implemented in the event that the CREVS becomes inoperable.

The NRC staff notes that the limiting condition for operation actions allow the redundant active CREVS components to be inoperable for 7 days. This is acceptable because these components are redundant with 100% capacity. The NRC staff also notes that in the event the single control room emergency ventilation system filter train or two recirculation fans or two recirculation dampers becomes inoperable, Action a.5 establishes adequate compensatory measures for this condition. The NRC staff finds this acceptable because it is consistent with the intent of NUREG-1431 "Standard Technical Specifications-Westinghouse Plants." In addition, the NRC staff finds that the requested changes are consistent with NUREG-0800 Revision 3 "Standard Review Plan" Section 9.4.1, "Control Room Area Ventilation System," acceptance criteria for the control room, in that the requested changes addresses the requirements of GDC 19 regarding the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents, and adequate protection against radiation and hazardous chemical releases are provided to permit access to and occupancy of the control room under accident conditions. Therefore, the NRC staff finds the requested changes to TS 3/4.7.5 acceptable.

3.5.6 TS 3.7.9 Gas Decay Tanks

The licensee has proposed an administrative change to the Gas Decay Tanks LCO clarifying the phrase within the parentheses from "(considered as Xe-133 equivalent)" to the newly defined "(DOSE EQUIVALENT XE-133)." This change is acceptable to the NRC staff since it is an administrative change to provide clarity and does not change the maximum curie content allowed in each gas decay tank.

3.5.7 TS 6.8.4h. Containment Leakage Rate Testing Program

The licensee has proposed to change the TS maximum allowable containment leakage limit, La, in TS 6.8.4.h, from 0.25% to 0.20% of containment air weight per day. The licensee reviewed the historical data from containment Integrated Leakage Rate Tests and performed an engineering evaluation to ensure that the change in the allowable limit acceptance criterion can be achieved. The licensee asserts that the proposed limit is more restrictive than the limit currently in place and that a maximum allowable leakage rate La of 0.20% supports the value assumed in the AST design basis LOCA and the RCCA Ejection event analyses to determine the dose consequences. Therefore, the NRC staff finds this proposed change in the TS maximum allowable containment leakage limit to be acceptable from a dose consequence perspective.

3.5.8 TS 6.8.4.j.b.2, Steam Generator Program

The licensee has proposed to revise the SG Program to reduce the accident induced primary to secondary leakage performance criterion from 1 gpm (1440 gpd) through all SGs and 500 gpd through one SG at accident conditions to 0.60 gpm (864 gpd) through all SGs and 0.20 gpm (288 gpd) through any one SG at room temperature conditions. The proposed limit of 288 gpd at room temperature conditions equates to approximately 398 gpd at accident conditions. Therefore, the proposed reduction in accident induced primary to secondary leakage is approximately 20%.

The licensee asserts that the proposed limits are lower than the limits currently in place and are consistent with the leakage rates assumed in the AST accident analyses, other than SGTR, in terms of total leakage rate for all SGs and leakage rate for an individual SG. The operational limit on primary to secondary leakage limit in any one SG is 150 gpd at room temperature. The proposed accident induced limit of 0.20 gpm (288 gpd) at room temperature conditions allows for an approximate 2 to 1 margin between the operational and the accident induced primary to secondary leakage limits.

TS 3.4.6.2, RCS Operational Leakage, limits the primary to secondary leakage through any one SG to 150 gpd and is unchanged by the AST analyses. This will have no impact on plant operations. The AST analyses conservatively assume the primary to secondary leakage to be maximized at 0.20 gpm (288 gpd) using a room temperature water density. Therefore, the described changes to TS 6.8.4.j.b.2, Steam Generator Program are consistent with the revised AST dose consequence analyses and are, therefore, acceptable to the NRC staff from a dose consequence perspective.

3.5.9 TS 6.9.1.2 Annual Reports

The licensee has proposed an administrative change to ensure that the reporting requirements agree with the applicable TS LCO and ACTION requirements. This change is necessary to reflect the decrease in the primary coolant specific activity from 1.0 microcurie per gram DOSE EQUIVALENT I-131 to 0.25 microcurie per gram DOSE EQUIVALENT I-131. This change is consistent with the revised AST dose consequence analyses and is, therefore, acceptable to the NRC staff.

3.5.10 TS 3/4.6.2.3 Recirculation pH Control System

The licensee has proposed a new TS to maintain the containment sump pool pH greater than 7.0. The minimum pH evaluation used the maximum borated water source volumes and concentrations and the contribution of acid from radiolysis of cables and sump fluid to determine the minimum sodium tetraborate mass needed to ensure an equilibrium sump pH greater than 7.0. The calculation determined that 11,061 lb of sodium tetraborate would be sufficient to maintain pH greater than 7.0. Any quantity of sodium tetraborate greater than 11,061 lb will ensure that the sump pool pH will remain in an alkaline regime under the worst case boron concentrations, sump fluid volumes, and quantities of strong acid generated. The proposed TS SRs ensure that at least once per 18 months the licensee verifies that the baskets are in place and that they collectively contain 11,061 pounds (227 cubic feet based on minimum density) of sodium tetraborate. The TS limit on minimum sodium tetraborate mass will ensure that there is sufficient sodium hydroxide available to maintain the post-LOCA sump pH above 7.0.

4.0 LICENSEE COMMITMENTS

The following commitments were submitted by the licensee:

1. FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR Part 50 Appendix A, GDC 19. The new intakes will be located near the ground level of the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for postaccident contaminants. FPL will perform

postmodification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes.

2. FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

3. FPL will implement the requirements of the proposed TS for the Recirculation pH Control System to maintain operability of this system and ensure the suitability of the NaTB. This commitment supersedes the commitment from PTN [Turkey Point] Letters L-2009-062 and L-2009-063 (References 28 and 29, respectively) related to NaTB sampling.

4. FPL will replace the aluminum fins on the normal containment coolers with copper fins. The copper fins will maintain the post-LOCA debris generation within the quantities currently assumed in the PTN sump strainer design basis.

5. FPL will update the necessary procedures to implement the manual operator action for initiation of the emergency CR ventilation system.

6. The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at PTN, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the PTN UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the PTN UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at PTN.

5.0 LICENSE CONDITIONS

The NRC staff's acceptance of an LAR cannot be based on assuming the fulfillment of licensee commitments. The relocation of the CR emergency air intakes, installation of NaTB baskets, and design of the CREVS compensatory filtration unit are necessary to ensure that the assumptions and design inputs described in the AST dose consequence analysis, as described in the LAR, remain valid. The NRC staff's acceptance of this AST LAR is based on the completion of actions identified in the LAR as commitments 1, 2, and 6, as stated in Section 4.0 of this SE. Therefore, the staff has concluded that its approval of the AST LAR is conditional upon the completion of these three activities prior to the implementation of the AST license amendment. Below are the license conditions:

- FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR Part 50 Appendix A, GDC 19. The new intakes will be located near the ground level extending out from the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes. In addition, FPL will provide to the NRC a confirmatory assessment which demonstrates that the requirements of 10 CFR 50 Appendix A, GDC 19 will be met. The confirmatory assessment will follow the methodology in Amendments 244 and 240 [the alternative source term amendment] including the methods used for the establishment of the atmospheric dispersion factors (X/Q values).

- FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

- The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey Point, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 as noticed (75 FR 39978). 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 the amendments.

7.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: John Parillo
Leta Brown
Jason White
Harold Walker
Matthew Yoder
Kenn Miller
William Jessup
Dan Hoang

Date: June 23, 2011

Attachments:
Tables 1 through 12

Table 1
Turkey Point Units 3 and 4, Radiological Consequences Expressed as TEDE [1]
(rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR [4]
LOCA	4.7	1.2	4.5
MSLB Pre-accident spike	0.023	0.018	1.6
SGTR Pre-accident spike	0.67	0.14	3.1
Dose acceptance criteria	25	25	5
MSLB Concurrent iodine spike	0.037	0.032	1.6
SGTR Concurrent iodine spike	0.24	0.052	1.3
Locked Rotor (Automatic CR isolation)	0.47	0.47	1.2
Locked Rotor (Manual CR isolation)	0.076	0.076	1.3
Dose acceptance criteria	2.5	2.5	5
FHA – Containment	0.73	0.15	1.3
FHA – Fuel Handling Building	0.73	0.15	3.9
Spent Fuel Cask Drop	0.32	0.064	2.1
RCCA Ejection Containment Release [5]	0.70	0.29	2.3
RCCA Ejection Secondary Side Release [5] (Automatic CR isolation)	0.49	0.43	1.1
RCCA Ejection Secondary Side Release [6] (Manual CR isolation)	0.29	0.26	3.4
Dose acceptance criteria	6.3	6.3	5
WGDT rupture	0.066	0.013	0.33
Dose acceptance criteria	0.1	0.1	5

[1] Total effective dose equivalent
[2] Exclusion area boundary- worst 2-hour dose
[3] Low population zone - integrated 30 day dose
[4] Assumed unfiltered CR inleakage 100 cfm
[5] Assumes 10% DNB and 0.25% FCM
[6] Assumes 6.22% DNB and 0.16% FCM

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

Table 2 (Page 1 of 6)
Turkey Point Units 3 and 4
CR Atmospheric Dispersion Factors (χ/Q Values)

A. Loss-of-Coolant Accident (LOCA): Containment 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	Unit 4 Western most Electrical Penetration / Normal CR intake	1.15×10^{-2}	---	---	---	---
During CR Recirculation	Unit 4 Emergency Escape Lock / SE emergency CR intake (1)	1.46×10^{-3}	1.06×10^{-3}	3.97×10^{-4}	3.14×10^{-4}	2.35×10^{-4}

B. Loss-of-Coolant Accident (LOCA): 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	N/A	---	---	---	---	---
During CR Recirculation	Unit 4 RWST / SE emergency CR intake (1)	1.96×10^{-3}	1.55×10^{-3}	6.52×10^{-4}	4.84×10^{-4}	3.79×10^{-4}

[1] The SE emergency CR Intake receptor location qualifies for the dual intake credit allowed by Section 3.3.2.2 of Reg. Guide 1.194. This credit is not applied to the values shown in Table 2; however, all values for this receptor location are reduced by a factor of 2 when applied in the event analyses.

Table 2 (Page 2 of 6)
Turkey Point Units 3 and 4
CR Atmospheric Dispersion Factors (χ/Q Values)

C. Loss-of-Coolant Accident (LOCA): Refueling Water Storage Tank (RWST) 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	N/A	---	---	---	---	---
During CR Recirculation	Unit 4 RWST / SE emergency CR intake (1)	1.96×10^{-3}	1.55×10^{-3}	6.52×10^{-4}	4.84×10^{-4}	3.79×10^{-4}

D. Loss-of-Coolant Accident (LOCA): Containment 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	Plant stack / Normal CR intake	1.86×10^{-3}	---	---	---	---
During CR Recirculation	Plant stack / SE emergency CR intake (1)	9.05×10^{-4}	7.62×10^{-4}	2.83×10^{-4}	2.14×10^{-4}	1.61×10^{-4}

Table 2 (Page 3 of 6)
Turkey Point Units 3 and 4
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	Unit 4 Personnel Hatch / Normal CR intake	1.04×10^{-2}	---	---	---	---
During CR Recirculation	Unit 4 Emergency Escape Lock / SE emergency CR intake (1)	1.46×10^{-3}	1.06×10^{-3}	3.97×10^{-4}	3.14×10^{-4}	2.35×10^{-4}

F. Fuel Handling Accident (FHA): Fuel Handling Building (FHB) Release & Spent Fuel Cask Drop Accident

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	Unit 4 Spent Fuel Building (NW corner) / Normal CR intake	2.36×10^{-3}	---	---	---	---
During CR Recirculation	Unit 4 Spent Fuel Building (SE corner) / SE emergency CR intake (1)	3.39×10^{-3}	2.77×10^{-3}	1.07×10^{-3}	8.40×10^{-4}	6.49×10^{-4}

Table 2 (Page 4 of 6)
Turkey Point Units 3 and 4
CR Atmospheric Dispersion Factors (χ/Q Values)

G. Main Steam Line Break (MSLB): Break 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	Unit 4 Main Steam Line Closest Point / Normal CR intake	1.59×10^{-2}	---	---	---	---
During CR Recirculation	Unit 4 Main Steam Line Closest Point / SE emergency CR intake (1)	7.37×10^{-4}	4.57×10^{-4}	1.88×10^{-4}	1.33×10^{-4}	8.67×10^{-5}

H. Main Steam Line Break (MSLB): MSSV/ADV 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	Unit 4 Closest MSSV ADV (2)/ Normal CR intake	1.37×10^{-2} (3)	---	---	---	---
During CR Recirculation	Unit 4 Closest MSSV ADV (2) / SE emergency CR intake (1)	6.88×10^{-4} (3)	4.39×10^{-4}	1.77×10^{-4}	1.28×10^{-4}	8.08×10^{-5}

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[2] The atmospheric dispersion factor corresponding to the limiting MSSV or ADV is used for each time period. No distinction is made between automatic steam relief from the MSSVs and controlled releases from the ADVs for radiological purposes.

[3] This release location meets the requirements for the plume rise credit described in Section 6 of Reg. Guide 1.194. The 0-2 hour values of 1.37×10^{-2} and 6.88×10^{-4} shown in Table 2 are reduced by a factor of 5 when used in the applicable event analyses.

Turkey Point Units 3 and 4

CR Atmospheric Dispersion Factors (χ/Q Values)

I. 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** Unit 4 SJAE/ Normal CR intake	6.61×10^{-2}	---	---	---	---
Prior to CR Isolation	**After Turbine Trip** Unit 4 Closest MSSV ADV (2) / Normal CR intake	1.37×10^{-2} (3)	---	---	---	---
During CR Recirculation	Unit 4 Closest MSSV ADV (2) / SE emergency CR intake (1)	6.88×10^{-4} (3)	4.39×10^{-4}	1.77×10^{-4}	1.28×10^{-4}	8.08×10^{-5}

J. 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	Unit 4 Closest MSSV ADV (2) / Normal CR intake*	1.37×10^{-2} (3)	---	---	---	---
During CR Recirculation	Unit 4 Closest MSSV ADV (2) / SE emergency CR intake (1)	6.88×10^{-4} (3)	4.39×10^{-4}	1.77×10^{-4}	1.28×10^{-4}	8.08×10^{-5}

Table 2 (Page 6 of 6)
Turkey Point Units 3 and 4
CR Atmospheric Dispersion Factors (χ/Q Values)

K. Rod Cluster Control Assembly (RCCA) Ejection: Containment Leakage

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	Unit 4 Westernmost Electrical Penetration / Normal CR intake	1.15×10^{-2}	---	---	---	---
During CR Recirculation	Unit 4 Emergency Escape Lock / SE emergency CR intake (1)	1.46×10^{-3}	1.06×10^{-3}	3.97×10^{-4}	3.14×10^{-4}	2.35×10^{-4}

L. Rod Cluster Control Assembly (RCCA) Ejection: Secondary Side Release

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	Unit 4 Closest MSSV ADV (2) / Normal CR intake	1.37×10^{-2} (3)	---	---	---	---
During CR Recirculation	Unit 4 Closest MSSV ADV (2) / SE emergency CR intake (1)	6.88×10^{-4} (3)	4.39×10^{-4}	1.77×10^{-4}	1.28×10^{-4}	8.08×10^{-5}

M. Waste Gas Decay Tank (WGDT) Rupture

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	Auxiliary Building Vent V-10 / Normal CR Intake	2.84×10^{-3}	2.58×10^{-3}	1.28×10^{-3}	1.19×10^{-3}	8.45×10^{-4}
During CR Recirculation	N/A	---	---	---	---	---

Table 3
Turkey Point Units 3 and 4
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**	$1.37 \times 10^{-4*}$	7.89×10^{-5}	6.00×10^{-5}	3.30×10^{-5}	1.40×10^{-5}
	LPZ	2.73×10^{-5}	1.23×10^{-5}	8.24×10^{-6}	3.46×10^{-6}	9.95×10^{-7}

*Note that *all* releases are assumed to be ground-level pursuant to RG. 1.145.

Table 4
Turkey Point Units 3 and 4, CR Data and Assumptions and Direct Shine Results

CR Volume	47,786 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	100 cfm
Emergency Operation	
Recirculation Mode:	
Filtered make-up flow rate	525 cfm
Filtered recirculation flow rate	375 cfm
Unfiltered Make-up Flow Rate	0 cfm
Assumed unfiltered inleakage	100 cfm
Filter Efficiencies	
Particulates	99%
Elemental iodine	95%
Organic iodine	95%
CR Assumed Breathing Rate	
0 - 720 hours	3.5×10^{-4} 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	
Source	Direct Shine Dose (rem)
Containment	0.059
Containment purge duct	0.333
CR filters	0.270
External cloud	0.061
Total	**0.723**

Table 5 (Page 1 of 3)
Turkey Point Units 3 and 4 Data and Assumptions for the LOCA

Core Power level	2652 MWt (2644 MWt + 0.3%)	
Core Average Fuel Burnup	45,000 MWD/MTU	
Fuel Enrichment	3.0 – 5.0 weight percent (w/o)	
Initial RCS Equilibrium Activity in coolant blowdown	0.25 □Ci/gm DEI 447.7 □Ci/gm DEX	
RCS Mass (maximum)	397,544 lbm	
Volumetric flow rate due to open purge valves	7,000 cfm	
Duration of flow through open purge valves	8 seconds	
Containment Purge Filtration	0%	
Total Containment Volume	1,550,000 ft^3	
Primary containment leak rate		
0 - 24 hours	0.2% (by weight)/day	
24 - 720 hours	0.1% (by weight)/day	
Surface Area for Wall Deposition	537,903 ft^2	
Elemental iodine wall deposition coefficient (0-720 hours)	5.58 hr^{-1}	
Particulate natural deposition removal coefficient	Unsprayed region	Sprayed region
0 - 720 hours	0.1 hr^{-1}	0 hr^{-1}
Spray Initiation Time	63.8 seconds (0.01772 hr)	
Containment sprayed region volume	534,442 ft^3	
Spray Fall Height	70 feet	
Containment upper unsprayed region volume	643,864 ft^3	
Containment lower unsprayed region volume	371,694 ft^3	
Flow rate between sprayed and upper unsprayed regions	990,000 cfm	
Flow rate between upper and lower unsprayed regions	375,000 cfm	
Elemental iodine spray removal coefficients:		
0.01772 – 2.305 hours	20 hr^{-1}	
2.305 - 720 hours	0 hr^{-1} DF = 200 @ 2.305 hr	
Particulate iodine spray removal coefficients		
0.01772 – 3.061 hours	6.44 hr^{-1}	
After 3.061 hours	0.644 hr^{-1} DF = 50 @ 3.061 hr	

Table 5 (Page 2 of 3)
Turkey Point Units 3 and 4 Data and Assumptions for the LOCA

Volume of water in containment sump (minimum)	239,000 gallons (31,949 ft^3)
ECCS Leakage (2 times allowed value)	4,650 cc/hr
ECCS Flashing fraction	
Calculated	0.092
Used for dose determination	0.10
Chemical form of released iodine from ECCS leakage	
Elemental	97%
Organic	3%

No credit for filtration or dilution for ECCS leakage - activity released directly to environment

Initial RWST liquid inventory (minimum)	60,000 gallons
ECCS leakage to RWST (2 times allowed value)	0.1 gph
Flashing fraction for leakage into RWST	0 %
RWST liquid/vapor elemental iodine partition factor	41.18
Time dependent RWST total iodine concentration (gm-atom/liter)	
Selected times in hours	RWST Iodine concentration
0.00	0.00E+00
8.0	8.660 $\times 10^{-10}$
24.0	2.654 $\times 10^{-9}$
96.0	1.070 $\times 10^{-8}$
720.0	8.033 $\times 10^{-8}$
Time dependent RWST elemental iodine fraction	
Selected times in hours	Elemental iodine fraction
0.0	0.00
8.0	1.143 $\times 10^{-3}$
24.0	3.485 $\times 10^{-3}$
96.0	1.376 $\times 10^{-2}$
720.0	8.822 $\times 10^{-2}$
LOCA Adjusted iodine release rate from RWST	
Time (hours)	Iodine release rate (cfm)
0.25	1.225 $\times 10^{-7}$
12.0	8.437 $\times 10^{-7}$
72.0	1.546 $\times 10^{-6}$
100.0	3.603 $\times 10^{-6}$
300.0	6.796 $\times 10^{-6}$
500.0	9.977 $\times 10^{-6}$
600.0	1.089 $\times 10^{-5}$
700.0	1.148 $\times 10^{-5}$
720.0	1.148 $\times 10^{-5}$

Table 5 (Page 3 of 3)
Turkey Point Units 3 and 4 Data and Assumptions for the LOCA

CR ventilation assumptions

CR isolation signal	High containment radiation
Time of CR isolation	30 seconds
Unfiltered inleakage	100 cfm

Transport assumptions:

Containment leakage release	Nearest containment penetration to CR intake
ECCS leakage	RWST vent (see note below)
RWST backleakage	RWST vent
Containment purge	Plant stack

Note: The licensee modeled the activity from ECCS components and from RWST leakage as an unfiltered ground level releases from the location of the RWST. For the ECCS leakage, the licensee has determined that the atmospheric dispersion factors from the RWST to the CR emergency intakes are more limiting than from any of the Auxiliary Building penetrations.

Table 6
Turkey Point Units 3 and 4 Data and Assumptions for the FHA

Core thermal power level	2652 MWt (2644 MWt + 0.3%)
Discharged fuel assembly burnup	45,000 MWD/MTU
Fuel enrichment	3.0 – 5.0 w/o
Maximum radial peaking factor	1.65
Number of fuel assemblies in the core	157
Number of fuel assemblies damaged	1
Minimum post shutdown fuel handling time (decay time)	72 hours
Minimum pool water depth above damaged fuel	23 feet
Fuel clad damage gap release fractions NUREG/CR-55009	
I-131	0.12
Remainder of halogens	0.10
Kr-85	0.30
Remainder of noble gases	0.10
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
CR ventilation assumptions	
Isolation time containment release	30 seconds
Isolation time FHB release	30 minutes for manual isolation
Assumed unfiltered inleakage	100 cfm

Table 7 (Page 1 of 2)
Turkey Point Units 3 and 4 Data and Assumptions for the MSLB Accident

Core Power level	2652 MWt (2644 + 0.3%)
Initial RCS Equilibrium Activity	
Iodine	0.25 □Ci/gm DEI
Noble gas	447.7 □Ci/gm DEX
Secondary coolant iodine activity	0.1 □Ci/gm DEI
Maximum pre-accident spike iodine concentration	60 μCi/gm DEI
Accident initiated iodine spike appearance rate	500 times equilibrium rate
Duration of accident initiated spike	8 hours
Primary to secondary leak rates	0.2 gpm/SG (0.6 gpm total)
RCS density for leak rate conversion to lbm/ft^3	62.4 lbm/ft^3
Time to establish shutdown cooling and terminate steam release	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
Reactor coolant system (RCS) mass	366,086 lbm (minimum)
SG secondary side mass	
Faulted SG	131,516.5 lbm
Intact SGs	67,707 lbm per SG
Faulted SG release rate	Instantaneous
Time to re-cover intact SG tubes	30 minutes
Assumed flashing fraction during tube uncovery period	11%
SG secondary side iodine partition	
Faulted SG	1(none)
Intact SG	100
Chemical form of iodine released from the secondary side	
Elemental	97%
Organic	3%

Intact SGs steam release rate in lbm/min for time interval in hours

0.0 (hours)	2622 (lbm/min)
2.0	2058
3.0	1931
4.0	1814
5.0	1694
8.0	1070
11.0	965
16.0	864
24.0	820
63.0	0.0

Table 7 (Page 2 of 2)
Turkey Point Units 3 and 4 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	30 minutes
Flashing fraction	11%
Iodine equilibrium appearance assumptions	
Letdown flow rate	132 gpm
Identified RCS leakage	10 gpm
Unidentified RCS leakage	1 gpm
RCS mass	397,544 lbm

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)
I-131	53.90	25,870
I-132	101.83	48,881
I-133	76.17	36,564
I-134	38.22	18,343
I-135	50.50	24,241

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

Isotope	Activity in μCi/gm
I-131	48.1440
I-132	34.1280
I-133	58.3440
I-134	6.3192
I-135	28.8000

CR ventilation assumptions	
Isolation time (total)	41.5 seconds
CR isolation on safety injection signal	11.5 seconds
Delay for DG start, fan start and dampers	30 seconds
After CR isolation	
Filtered makeup flow	525 cfm
Filtered recirculation flow	375 cfm
Assumed unfiltered inleakage	100 cfm

Table 8 (Page 1 of 2)
Turkey Point Units 3 and 4 Data and Assumptions for the SGTR Accident

Core power level	2652 MWt (2644 MWt + 0.3%)
Initial RCS Equilibrium Activity	
Iodine	0.25 □Ci/gm DEI
Noble gas	447.7 □Ci/gm DEX
Initial secondary side equilibrium activity	0.1 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
Time of reactor trip	291 seconds (0.0808 hrs)
Break flow flashing fraction	
Prior to reactor trip	21%
Following reactor trip	11%
Time to isolate ruptured SG and terminate break flow	30 minutes
Primary to secondary SG tube leakage rate	0.2 gpm per SG
RCS density based on leakage test conditions	62.4 lbm/ft^3
Time to establish shutdown cooling and terminate steam release from intact SGs	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
RCS mass (minimum)	366,086 lbm
Secondary coolant system mass	67,707 lbm per SG
Time to recover intact SG tubes	30 minutes
Tube uncovery flashing fraction	11%
SG secondary side iodine partition coefficients	
Flashed tube flow	None
Non-flashed tube flow	100
Condenser	100

SGTR mass flow rates in lbm/min during time period in hours used for Dose Analysis

			Steam Release to Atmosphere	
Event @ Initial Time	Time (Hours)	Break flow	Ruptured SG (RSG)	Intact SG
SGTR	0 to 0.0808	6507	64,800	129,600
Rx Trip & LOOP	0.0808 – 0.5	4161	3579	4033
Break flow terminated RSG isolated	0.5 – 2	0	0	4033
	2 – 8	0	0	2833
	8 – 24	0	0	1525
	24 – 63	0	0	1270
RHR provides cooldown Steam release terminated	63 - 720	0	0	0

Table 8 (Page 2 of 2)
Turkey Point Units 3 and 4 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	36.11	17,333
I-132	68.23	32,750
I-133	51.04	24,498
I-134	25.60	12,290
I-135	33.84	16,241

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

Isotope	Activity in μCi/gm
I-131	48.1440
I-132	34.1280
I-133	58.3440
I-134	6.3192
I-135	28.8000

Chemical form of iodine released from SGs

Particulate	0 %
Elemental	97%
Organic	3%

CR ventilation assumptions

Isolation time (total)	321 seconds
CR isolation on safety injection signal	291 seconds
Delay for DG start, fan start and dampers	30 seconds
After CR isolation	
Filtered makeup flow	525 cfm
Filtered recirculation flow	375 cfm
Assumed unfiltered inleakage	100 cfm

Table 9
Turkey Point Units 3 and 4 Data and Assumptions for the Locked Rotor Accident

Core power level	2652 MWt (2644 MWt + 0.3%)
Core average fuel burnup	45,000 MWD/MTU
Fuel enrichment	3.0 – 5.0 weight percent (w/o)
Radial peaking factor	1.65
Initial secondary side equilibrium activity	0.1 μCi/gm DEI
Percent of fuel rods in DNB	15%
RCS mass – minimum used to maximize dose	366,086 lbm
Total primary to secondary leak rate from SGs	0.2 gpm per SG
Time to establish shutdown cooling to terminate steam release	63 hours
Time for RCS to reach 212°F to terminate SG tube leakage	125.4 hours
Time to recover intact SG tubes	30 minutes
Tube uncovery flashing fraction	11%
Secondary coolant system mass	67,707 lbm per SG
RCS density based on leakage test conditions	62.4 lbm/ft^3
SG secondary side iodine partition coefficients	
Flashed tube leakage	1(none)
Non-flashed tube leakage	100

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

Event	Time (Hours)	SG release rate (lbm/min)
LRA	0.00 – 2.0	2598
	2.0 – 3.0	2143
	3.0 – 4.0	2016
	4.0 – 5.0	1900
	5.0 – 8.0	1779
	8.0 – 11.0	2598
	11.0 – 16.0	965
	16.0 – 24.0	864
	24.0 – 63.0	820
	63.0 – 720	0.0

CR ventilation assumptions

CR isolation signal (Automatic)	CR intake high radiation
Isolation time (Automatic)	60 seconds
Isolation time (Manual operator action)	30 minutes
Assumed unfiltered inleakage	100 cfm

Table 10 (Page 1 of 2)
Turkey Point Units 3 and 4 Data and Assumptions for the RCCA Ejection Accident

Core Power level	2652 MWt (2644 MWt + 0.3%)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 5.0 weight percent (w/o)
Maximum radial peaking factor	1.65
Percent of fuel rods in DNB	
Design basis case	10%
Manual CR isolation case	6.22%
Percent of fuel rods with FCM	
Design basis case	0.25%
Manual CR isolation case	0.16%
Initial secondary side equilibrium activity	0.1 μCi/gm DEI
SG secondary side iodine partition coefficients	
Flashed tube leakage	1(none)
Non-flashed tube leakage	100
SG tube leakage rate	0.2 gpm per SG
Time to establish shutdown cooling and terminate steam release	30 minutes
Time to recover SG tubes	30 minutes
Tube uncovery flashing fraction	11%
RCS mass – minimum used to maximize dose	366,086 lbm
Secondary coolant system mass	67,707 lbm per 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%
CR isolation assumptions	
CR Isolation – containment release	30 sec – containment monitors
CR isolation – secondary (automatic)	60 sec – CR intake monitors
CR isolation – secondary (manual)	30 minutes – manual isolation
Assumed unfiltered inleakage	100 cfm

Table 10 (Page 2 of 2)
Turkey Point Units 3 and 4 Data and Assumptions for the RCCA Ejection Accident

Containment volume	1,550,000 ft^3
Containment leakage rate	
0 to 24 hours	0.2% (by weight)/day
24 – 720 hours	0.1% (by weight)/day
Containment natural deposition coefficients	
Aerosols	0.1 hr^{-1}
Elemental iodine	5.58 hr^{-1}
Organic iodine	None
Credit for containment sprays	None

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

Event	Time (Hours)	SG release rate (lbm/min)
RCCA Ejection	0.00 – 2.0	2598
	2.0 – 3.0	2143
	3.0 – 4.0	2016
	4.0 – 5.0	1900
	5.0 – 8.0	1779
	8.0 – 11.0	2598
	11.0 – 16.0	965
	16.0 – 24.0	864
	24.0 – 63.0	820
	63.0 – 720	0.0

Table 11
Turkey Point Units 3 and 4 Data and Assumptions
Waste Gas Decay Tank (WGDT) Rupture

Core Power level	2652 MWt (2644 MWt + 0.3%)
WGDT inventory	84,274.8 Curies DEX
WGDT volume	525 ft^3
Arbitrary tank leak rate to model rupture	1 $x10^{+6}$ cfm

CR ventilation assumptions

CR isolation	Not isolated
Unfiltered makeup flow	1000 cfm
Assumed unfiltered inleakage	100 cfm

WGDT source term

Isotope	Tank inventory (Curies)
Kr-85m	214.22
Kr-85	6286.02
Kr-87	129.06
Kr-88	381.02
Xe-131m	513.74
Xe-133	42555.99
Xe-133m	590.55
Xe-135	916.04
Xe-135m	82.41
Xe-138	85.94

Table 12
Turkey Point Units 3 and 4 Data and Assumptions for the Spent Fuel Cask Drop

Core thermal power level	2652 MWt (2644 MWt + 0.3%)
Core average fuel burnup	45,000 MWD/MTU
Fuel enrichment	3.0 – 5.0 w/o
Number of fuel assemblies damaged	157
Delay before cask movement (decay time)	1525 hours
Minimum pool water depth above damaged fuel	23 feet
Fuel clad damage gap release fractions NUREG/CR-55009	
I-131	0.12
Remainder of halogens	0.10
Kr-85	0.30
Remainder of noble gases	0.10
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
CR ventilation assumptions	
Isolation time	30 minutes for manual isolation
Assumed unfiltered inleakage	100 cfm

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/

Jason C. Paige, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 244 to DPR-31
2. Amendment No. 240 to DPR-41
3. Safety Evaluation

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