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

November 14, 2018

Mr. Mano Nazar
President, Nuclear Division
and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATIONS CHANGES TO ADDRESS NON-CONSERVATIVE ACTIONS FOR CONTAINMENT AND CONTROL ROOM VENTILATION FUNCTIONS (EPID L-2017-LLA-0425)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 283 to Renewed Facility Operating License No. DPR-31 and Amendment No. 277 to Renewed Facility Operating License No. DPR-41 for Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively. The amendments change the Technical Specifications (TSs) in response to the application from Florida Power & Light Company dated December 21, 2017, as supplemented by letter dated June 12, 2018.

The amendments revise the TSs pertaining to the Engineered Safety Features Actuation System instrumentation to resolve non-conservative actions associated with the Containment ventilation isolation and the Control Room ventilation isolation functions. In addition, the amendments revise the Control Room ventilation isolation function to no longer credit Containment radiation monitoring instrumentation, eliminate redundant radiation monitoring instrumentation requirements, eliminate select core alterations applicability requirements, relocate radiation monitoring and Reactor Coolant System leakage detection requirements within the TS to align with their respective functions, and relocate the Spent Fuel Pool area monitoring requirements to licensee-controlled documents. The NRC staff's safety evaluation of the amendments is enclosed.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Michael J. Wentzel, 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. 283 to DPR-31
2. Amendment No. 277 to DPR-41
3. Safety Evaluation

cc: Listserv

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 283
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated December 21, 2017, as supplemented by letter dated June 12, 2018, 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 Renewed Facility 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. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 283, 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. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: November 14, 2018

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 277
Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated December 21, 2017, as supplemented by letter dated June 12, 2018, 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 Renewed Facility 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. 277, 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. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop
Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: November 14, 2018

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 283 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 277 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

Replace page 3 of Renewed Facility Operating License No. DPR-31 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace page 3 of Renewed Facility Operating License No. DPR-41 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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 areas of change.

Remove	Insert
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
------	3/4 3-22A
3/4 3-25	3/4 3-25
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42	3/4 3-42
3/4 3-43	3/4 3-43
3/4 4-13	3/4 4-13
3/4 9-9	3/4 9-9
3/4 9-12	3/4 9-12

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 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 283, 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.

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 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, 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.

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Ventilation Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4**	24A, 24B
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Deleted					
d. Containment Isolation Manual Phase A or Manual Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24A, 24B

TABLE 3.3-2 (Continued)

TABLE NOTATION

Trip function may be blocked in this MODE below the Pressurizer Pressure Interlock Setpoint of 2000 psig.

Channels are for particulate radioactivity and for gaseous radioactivity. Either an OPERABLE particulate radioactivity or gaseous radioactivity channel will satisfy the Minimum Channels OPERABLE requirement.

Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

####Steam Generator overfill protection is not part of the Engineered Safety Features Actuation System (ESFAS), and is added to the Technical Specifications only in accordance with NRC Generic Letter 89-19.

* Trip function may be blocked in this MODE below the T_{avg}--Low Interlock Setpoint.

** Only during movement of irradiated fuel within the containment.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 16 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE Requirement, operation may continue provided the Containment purge supply, exhaust and instrument air bleed valves are maintained closed. (The instrument air bleed valves may be opened intermittently under administrative controls).

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. Both channels of any one load center may be taken out of service for up to 8 hours in order to perform surveillance testing per Specification 4.3.2.1.

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement:

(a) Restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

(b) Restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 24A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 7 days restore the inoperable channel to OPERABLE status or place the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 24B - With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement, either:

1. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with BOTH Control Room emergency recirculation fans operating, OR

2. a. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with ONE Control Room emergency recirculating fan operating, AND

   b. Restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both Units simultaneously, then be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 25 - With number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
3. Containment Isolation (Continued)		
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	≤22.6 psig	≤20.0 psig
Coincident with: Containment Pressure--High	≤4.5 psig	≤4.0 psig
c. Containment Ventilation Isolation		
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
4) Containment Radioactivity--High	Particulate (R-11) $\leq 6.8 \times 10^5$ CPM Gaseous (R-12) See Note 2	Particulate (R-11) $\leq 6.1 \times 10^5$ CPM Gaseous (R-12) See Note 2
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
8. Engineering Safety Features Actuation System Interlocks		
a. Pressurizer Pressure	≤2018 psig	Nominal 2000 psig
b. Tavg--Low	≥542.5°F	Nominal 543°F
9. Control Room Ventilation Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
c. Deleted		
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.
e. Air Intake Radiation Level	≤2.83 mR/hr	≤2 mR/hr

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

(1) Deleted

(2) Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

Containment Gaseous Monitor Allowable Value = $\frac{(3.5 \times 10^4)}{(\quad F \quad)}$ CPM,

Where F = $\frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

(3) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

(4) Time constants utilized in lead-lag controller for Steam Generator Pressure-Low and Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

If no Allowable Value is specified, as indicated by [ ], the trip setpoint shall also be the allowable value.

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Engineering Safety Features Actuation System Interlocks						
a. Pressurizer Pressure	N.A.	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)
b. Tavg--Low	N.A.	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)
9. Control Room Ventilation Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(4)
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Deleted						
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3, 4
e. Control Room Air Intake Radiation Level	SFCP	SFCP	SFCP	N.A.	N.A.	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

In accordance with the Surveillance Frequency Control Program each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in UFSAR Section 7.2

(1) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

(3) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.

(4) Applicable in MODES 1, 2, 3, 4 or during movement of irradiated fuel within the containment.

(5) Test of alarm function not required when alarm locked in.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $\leq 6.1 \times 10^5$ CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 for MODES 5 and 6

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During movement of irradiated fuel within the containment, comply with Specification 3/4.9.13.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

$$\text{Where } F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, comply with the following:

1) Table 3.3-2, ACTION 16, and

2) Technical Specification 3.4.6.1, ACTION a.

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - In MODES 5 or 6 (except during or movement of irradiated fuel within the containment): With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement perform the following:

1) Obtain and analyze appropriate grab samples at least once per 24 hours, and

2) Monitor containment atmosphere with area radiation monitors.

Otherwise, isolate all penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

During movement of irradiated fuel within the containment: With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements, comply with ACTION statement requirements of Specification 3.9.9 and 3.9.13.

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION FOR PLANT
OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity--High	SFCP	SFCP	SFCP	All

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment Sump Level Monitoring System is OPERABLE;

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;

3) A Reactor Coolant System water inventory balance is performed at least once per 8* hours except when operating in shutdown cooling mode.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program, and

b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

* Not required to be performed until 12 hours after establishment of steady state operation.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the containment ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of irradiated fuel inside the containment by verifying that Containment Ventilation Isolation occurs on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.13 RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.9.13 The Containment Radiation monitors which initiate containment ventilation isolation shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

a) With one or both radiation monitors inoperable, operation may continue provided the containment ventilation isolation valves are maintained closed.

SURVEILLANCE REQUIREMENTS

4.9.13 Each Containment Radiation monitor shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 283 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 277 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated December 21, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17355A184), as supplemented by letter dated June 12, 2018 (ADAMS Accession No. ML18179A166), Florida Power & Light Company (FPL or the licensee) requested changes to the Technical Specifications (TSs) for Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point 3 and 4), which are contained in Appendix A of Renewed Facility Operating License Nos. DPR-31 and DPR-41. The licensee proposed to revise the TSs pertaining to the Engineered Safety Features Actuation System (ESFAS) instrumentation to resolve non-conservative actions associated with the Containment ventilation isolation and the Control Room ventilation isolation functions. In addition, the licensee proposed to revise the Control Room ventilation isolation function to no longer credit Containment radiation monitoring instrumentation, eliminate redundant radiation monitoring instrumentation requirements, eliminate select core alterations applicability requirements, relocate radiation monitoring and Reactor Coolant System leakage detection requirements within the TS to align with their respective functions, and relocate the Spent Fuel Pool area monitoring requirements to licensee-controlled documents.

By electronic mail (e-mail) dated May 16, 2018 (ADAMS Accession No. ML18136A724), the U.S. Nuclear Regulatory Commission (NRC) staff sent the licensee a request for additional information (RAI). From May 23 through June 7, 2018, the NRC staff conducted an audit via an online reference portal provided by the licensee. The staff performed the audit to ascertain the information needed to support its review of the application and develop RAIs, as needed. On July 13, 2018 (ADAMS Accession No. ML18177A305), the NRC staff issued an audit summary. The licensee responded to the RAI by letter dated June 12, 2018. The licensee's RAI response letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration (NSHC) determination that was published in the *Federal Register* (FR) on February 27, 2018 (83 FR 8516).

2.0 REGULATORY EVALUATION

2.1. System Descriptions

Engineered Safety Features Actuation System

The ESFAS both actuates the engineered safety features in the event that requisite accident signals are received and monitors their operation, once activated. Accident signals are generated by instrumentation that monitors temperatures, pressures, flows, and levels in the reactor coolant, steam, reactor containment and auxiliary systems. Process variables that are necessary on a continuous basis for safe and orderly operation of all systems and processes during the startup, operation, and shutdown of the plant are indicated, recorded and controlled from the Control Room.

The ESFAS actuates the engineered safety features by the respective actuation channels. The channels are designed such that a single failure will not defeat the specified function. This is done through the combination of redundant sensors, independent channel circuitry, coincident trip logic and different parameter measurements. Each coincidence network energizes an ESFAS actuation device that operates the associated engineered safety features equipment, motor starters and valve operators. Depending on the condition, the ESFAS may actuate several engineered safety features, including the Safety Injection, Containment ventilation isolation, and Control Room ventilation isolation systems.

Radiation Monitoring System

The Turkey Point 3 and 4 Radiation Monitoring System is designed to warn of radiation health violations that might develop and give early warnings of a malfunction that might lead to an unsafe health condition or plant damage. Instruments are located in and around Turkey Point 3 and 4 to detect and record radiation levels and to initiate alarms in the Control Room, should radiation levels exceed setpoints. The Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by plant staff to provide adequate information and warning for the continued safe operation of Turkey Point 3 and 4 and to provide assurance that personnel exposure will not exceed the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 20.

The Containment Air Particulate Monitors, R-3-11 (Unit 3) and R-4-11 (Unit 4) (R-11), measure the air particulate radioactivity levels in each containment through the continuous sampling of the containment atmosphere. The detector outputs are then transmitted to the Control Room radiation monitoring system cabinets. The Containment Air Particulate Monitors ensure that the radiation release rate through each containment vent during purging is maintained below specified limits. Upon the detection of high air-particulate-radioactivity, the Containment Air Particulate Monitors initiate closure of the containment purge supply and exhaust isolation valves and the containment instrument air bleed valves. Additionally, the Containment Air Particulate Monitors initiate isolation of the Control Room ventilation system. The alarm setpoints are specified in Table 3.3-3 of TS 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation," and set in accordance with the Turkey Point Offsite Dose Calculation Manual (ODCM).

The Containment Radioactive Gas Monitors, R-3-12 (Unit 3) and R-4-12 (Unit 4) (R-12), measure the gaseous beta radioactivity in each containment to ensure that the radiation release rate during containment purging is maintained below specified limits. The Containment

Radioactive Gas Monitors take continuous air samples from the containment atmosphere after passing through the Containment Air Particulate Monitors, R-11, draw the samples through a closed, sealed system to a gas monitor assembly, and transmit the detector outputs to the Control Room radiation monitoring system cabinets. High gas-radiation-level initiates closure of the containment purge supply and exhaust isolation valves and the containment instrument air bleed valves, and initiates isolation of the Control Room ventilation system. The alarm setpoints are specified in Table 3.3-3 of TS 3/4.3.2 and set in accordance with the Turkey Point ODCM.

The Unit 3 spent fuel pool area is monitored by the Spent Fuel Pit Vent Exhaust special particulate, iodine and noble gas (SPING) radiation monitor, RaD-3-6418, and the area radiation monitor, RD-3-1419. The Unit 3 Spent Fuel Pit Vent Exhaust SPING monitor detects gaseous radiation in the Unit 3 spent fuel pool and the Unit 3 cask handling facility areas. The Unit 3 Spent Fuel Pit Vent Exhaust SPING monitor also collects halogens and particulates on filter elements for later analyses by plant staff. The Unit 3 Spent Fuel Pit Vent Exhaust SPING monitor alarms on a control console in the Cable Spreading Room. The alarm setpoint is specified in Table 3.3-4 of TS 3/4.3.3, "Monitoring Instrumentation – Radiation Monitoring for Plant Operations."

Plant Vent Stack Exhaust SPING monitor, RaD-6304 and the Process Radiation Monitor, R-14, provide monitoring of the Unit 4 spent fuel pool area. Gaseous radiation in the Unit 4 spent fuel pool area is routed through the plant vent exhaust pathway for monitoring by both monitors. The Plant Vent Stack Exhaust SPING monitor also collect halogens and particulates on filter elements for later analyses by plant staff. The Process Radiation Monitor, R-14, detects gaseous radiation passing through the plant vent to the atmosphere and provides remote indication and annunciation on the Waste Disposal System control board in the Control Room. The Plant Vent Exhaust SPING monitor alarms on a control console in the Computer Room. A high radiation level alarm on R-14 automatically closes a gas release valve in the Waste Disposal System, thereby isolating the Gas Decay Tanks from the plant vent stack exhaust pathway. Both the RaD-6304 and R-14 alarm setpoints are specified in Table 3.3-4 of TS 3/4.3.3 and set in accordance with the Turkey Point ODCM.

Reactor Coolant System Leakage Detection System

The Reactor Coolant System (RCS) leakage detection systems are designed to identify the presence of significant leakage from the reactor coolant loops by monitoring radioactivity concentration in the Containment atmosphere, Auxiliary Building ventilation exhausts, Steam Generator blowdown, Condenser air ejector exhausts, and the component cooling loop liquid. Indications are provided to operators in the Control Room. RCS leakage detection equipment includes the Containment air particulate monitor, Containment radioactive gas monitor, the component cooling radiation monitor, and the Reactor Vessel Head leakage detection system, which is capable of sampling and analyzing each Control Rod Drive Mechanism Cooler Ventilation discharge and the Containment atmosphere on an as-needed basis. The Steam Generator blowdown and Condenser air ejector monitors function to detect primary-to-secondary system leakage. The basic design criterion is the detection of deviations from normal containment environmental conditions, including air particulate activity, radiogas activity, and, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Containment Purge System

The Containment Purge System is designed to purge the Containment atmosphere for unlimited access during shutdown periods. The Containment Purge System handles both supply and exhaust air. The supply system includes an outside air connection to roughing filters, a fan duct system, and a supply penetration with two quick-closing butterfly valves. The exhaust system includes an exhaust penetration, a duct system, fan, roughing filters with connection to the plant vent, and two quick-closing butterfly valves. The supply and exhaust penetration butterfly valves are in-series—one inside and one outside the containment—and act as purge isolation valves capable of closing in less than 5 seconds on receipt of a containment-isolation signal or high-activity signal from the air particulate and gas monitor.

Control Room Ventilation System

The Control Room Ventilation System, which includes the Control Room Emergency Ventilation System (CREVS), functions to provide a controlled environment for the comfort and safety of Control Room personnel and to assure the operability of Control Room components during normal operation, anticipated operational occurrences, and design-basis accident conditions. The design basis of the control room ventilation system for radiological emergencies is to be capable of automatically starting under accident conditions to initiate emergency control room pressurization and filtration, assuming the occurrence of a single active damper or supply fan failure. The design basis of the system with respect to other emergencies affecting the control room environment is to be capable of manual actuation.

The Control Room Ventilation System draws fresh air from the outside, but also has the capability to go into an emergency recirculation mode as part of the CREVS. During this mode, the normal outside air intake to the air handling units is closed and a separate dual CREVS intake is opened. To maintain an acceptable Control Room environment while in the emergency recirculation mode, fresh air provided from the CREVS intake piping and recirculated air from the Control Room is processed through the high efficiency particulate air filters, charcoal filters (single-train filtration unit), and supply fans. A manually aligned compensatory filtration unit is available as a backup to the CREVS filter train.

The Control Room emergency ventilation mode is initiated by a safety injection signal, a Phase "A" containment isolation signal, a high radiation signal from the Containment atmosphere radiation monitors, a high radiation signal from the redundant Control Room normal air intake radiation monitors, or manual initiation from a test switch. Following initiation, the Control Room ventilation exhaust fans are de-energized and the Control Room normal air intake and exhaust redundant, in-series ventilation isolation dampers are closed. The redundant, in-parallel Control Room emergency air intake dampers are opened to provide the emergency recirculation air path and a single air-supply fan is energized to move the appropriate mixture of recirculating Control Room air and fresh outdoor air through the CREVS filters. In addition, the Technical Support Center ventilation system emergency mode is started from the CREVS B channel. This feature is not part of the ESFAS nor required for the CREVS to meet its safety function.

2.2 Licensee's Proposed Changes

2.2.1 Changes to TS 3/4.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation

The licensee proposes to make the following modifications to Tables 3.3-2, 3.3-3, and 4.3-2 (deletions shown in stricken text and additions underlined):

Table 3.3-2

- Functional Unit (FU) 9.a:

APPLICABLE MODES	ACTION
1, 2, 3, 4~~, 6~~**	~~16~~ 24A, 24B

- FU 9.c would be deleted entirely.

- FU 9.e:

ACTION
~~16~~ 24A, 24B

- Table Notation # #:

Channels are for particulate radioactivity and for gaseous radioactivity. Either an OPERABLE particulate radioactivity or gaseous radioactivity channel will satisfy the Minimum Channels OPERABLE requirement.

- Table Notation **:

** Only during ~~CORE ALTERATIONS or~~ movement of irradiated fuel within containment.

- ACTION 16:

ACTION 16 - ~~With less than the Minimum Channels OPERABLE requirement, comply with the ACTION statement requirements of Specification 3.3.3.1 Item 1a of Table 3.3-4.~~ With the number of OPERABLE channels less than the Minimum Channels OPERABLE Requirement, operation may continue provided the Containment purge supply, exhaust and instrument air bleed valves are maintained closed. (The instrument air bleed valves may be opened intermittently under administrative controls).

- ACTION 24:

ACTION 24A - With the number of OPERABLE channels one less than the minimum Channels OPERABLE requirement, within ~~1 hour isolate the control room Emergency Ventilation System and initiate operation of~~7 days restore the inoperable channel to OPERABLE status or place the Control Room Emergency Ventilation System in the recirculation mode.

- Create new ACTION 24B:

ACTION 24B - With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement, either:

1. Immediately place both the Control Room Emergency Ventilation System in the recirculation mode with BOTH Control Room emergency recirculation fans operating, OR

2. a. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with ONE Control Room emergency recirculation fan operating, AND

   b. Restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both Units simultaneously, then be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours

Table 3.3-3

- FU 3.c.4:

4) Containment Radioactivity—High~~(1)~~

- FU 9.c would be deleted entirely.

- Table Notation 1:

(1) Deleted ~~Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.~~

Table 4.3-2

- FU 9.a:

  MODES FOR
  WHICH
  SURVEILLANCE
  IS REQUIRED
  (4)

- FU 9.c would be deleted entirely.

- Table Notation 4:

  (4) Applicable in MODES 1, 2, 3, 4 or during ~~CORE ALTERATIONS or~~ movement of irradiated fuel within the containment.

2.2.2 Changes to TS 3/4.3.3.1, "Radiation Monitoring for Plant Operations"

The licensee proposes to make the following modifications to Tables 3.3-4 and 4.3-3 (deletions shown in stricken text and additions underlined):

Table 3.3-4

- FU 1.b would be deleted entirely.

- FU 2.a would be deleted entirely.

- FU 2.b would be deleted entirely.

- Table Notation *:

  * During ~~CORE ALTERATIONS or~~ movement of irradiated fuel within the containment, comply with Specification 3/4.9.13.

- Table Notation **:

  ~~** With irradiated fuel in the spent fuel pits.~~

- Table Notation #:

  ~~# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.~~

- ACTION 26:

  ACTION 26 - In MODES 1 thru 4: With both Particulate and Gaseous Monitoring Systems inoperable, comply with the following: ~~operation may continue for up to 7 days provided:~~

1) Table 3.3-2, ACTION 16, and ~~A Containment sump level monitoring system is OPERABLE,~~

2) Technical Specification 3.4.6.1, ACTION a. ~~Appropriate grab samples are obtained and analyzed at least one per 24 hours,~~

~~3) A Reactor Coolant System water inventory balance is performed at least once per 8*** hours except when operating in shutdown cooling mode, and~~

~~4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.****~~

~~Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).~~

~~*** Not required to be performed until 12 hours after establishment of steady state operation.~~

~~**** Instrument Air Bleed Valves may be opened intermittently under administrative controls.~~

- ACTION 27:

ACTION 27 - In MODES 5 or 6 (except during ~~CORE ALTERATION or~~ movement of irradiated fuel within the containment): With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement perform the following:

1) Obtain and analyze appropriate grab samples at least once per 24 hours, and

2) Monitor containment atmosphere with area radiation monitors.

Otherwise, isolate all penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

During ~~CORE ALTERATION or~~ movement of irradiated fuel within the containment: With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements, comply with ACTION statement requirements of Specification 3.9.9 and 3.9.13.

- ACTION 28:

~~ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, immediately suspend operations in the Spent Fuel Pool area involving spent fuel manipulations.~~

Table 4.3-3

- FU 1.b.1 would be deleted entirely.

- FU 1.b.2 would be deleted entirely.
- FU 2.a would be deleted entirely.
- FU 2.b would be deleted entirely.
- Table Notation *:

  ~~* With irradiated fuel in the fuel storage pool areas.~~

- Table Notation #:

  ~~# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.~~

2.2.3 <u>Changes to TS 3/4.4.6, "Reactor Coolant System Leakage"</u>

- Limiting Condition for Operation (LCO) 3.4.6.1, ACTION a:

  a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

     1) A Containment Sump Level Monitoring System is OPERABLE;

     2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;

     3) A Reactor Coolant System water inventory balance is performed at least once per 8* hours except when operating in shutdown cooling mode<u>.</u> ~~; and~~

     ~~4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.**~~

  Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

- Surveillance Requirement (SR) 4.4.6.1:

  4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

  a. Containment Atmosphere Gaseous and Particulate Monitoring System performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST <u>in accordance with the Surveillance Frequency Control Program</u> ~~at the frequencies specified in Table 4.3-3~~, and

  b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

- Footnote **:

  ~~** Instrument Air Bleed valves may be opened intermittently under administrative controls.~~

2.2.4 Changes to TS 3/4.9.9, "Refueling Operations, Containment Ventilation Isolation Systems"

- LCO 3.9.9, APPLICABILITY:

  APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of irradiated fuel within the containment.

- SR 4.9.9:

  4.9.9 The Containment Ventilation Isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of irradiated fuel inside the containment ~~CORE ALTERATIONS~~ by verifying that Containment Ventilation Isolation occurs on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

2.2.5 Changes to TS 3/4.9.13, "Refueling Operations, Radiation Monitoring"

- LCO 3.9.13:

  3.9.13 The Containment Radiation monitors which initiate containment ~~and control room~~ ventilation isolation shall be OPERABLE.

  APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of irradiated fuel within the containment.

  ACTION:

  a) With one or both radiation monitors inoperable, operation may continue provided the containment ventilation isolation valves are maintained closed.

  ~~b) With one or both radiation monitors inoperable, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.~~

2.3 Regulatory Review

The NRC staff reviewed the licensee's application to determine whether (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that the activities proposed 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 the health and safety of the public. The NRC staff considered the following regulatory requirements, guidance, and licensing and design-basis information during its review of the proposed changes.

Paragraph 50.36(a)(1) of 10 CFR states, in part, that each applicant for an operating license shall include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36, "Technical specifications." Paragraph 50.36(c) of 10 CFR requires that the TSs include items in the following categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. Paragraph 50.36(c)(2) states, in part, that when an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

Paragraph (b)(2) of 10 CFR 50.67, "Accident source term," states:

> The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:
>
> (i) 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 Sv (25 rem)[1] total effective dose equivalent (TEDE).
>
> (ii) 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) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Turkey Point 3 and 4 were 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 3 and 4 are not licensed to the current GDC of 10 CFR Part 50, Appendix A. Section 1.3 of the Turkey Point 3 and 4 Updated Final Safety Analysis Report (UFSAR) provides a summary of the 1967 GDC proposed by the U.S. Atomic Energy Commission, as amended by the Atomic Industrial Forum (circa October 2, 1967). The NRC staff considered the following proposed GDC as part of its review:

- 1967 Proposed GDC 12, states that instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

- 1967 Proposed GDC 15 states that protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

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[1] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

- 1967 Proposed GDC 16 states that means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.
- 1967 Proposed GDC 17 states that means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity released to the environs of the plant have not been excessive.
- 1967 Proposed GDC 53 states that penetrations that require closure for the containment functions shall be protected by redundant valving and associated apparatus.

The licensee indicates throughout the Turkey Point 3 and 4 UFSAR that it is committed to continued compliance with the proposed GDC to which it was licensed in 1967, with the exception of the control room. The licensee has committed to 10 CFR Part 50, Appendix A, "Criterion 19 - Control room," as part of the change to the Alternative Source Term (AST) methodology for dose analysis as approved in License Amendment Nos. 244 (Unit 3) and 240 (Unit 4), dated June 23, 2011 (ADAMS Accession No. ML110800666).

Appendix A to 10 CFR Part 50, GDC Criterion 19 - Control room, states, in part:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

License Amendment Nos. 244 (Unit 3) and 240 (Unit 4) used an AST methodology for analyzing the radiological consequences of eight design-basis accidents using Regulatory Guide (RG) 1.183. The regulatory requirements and guidance applicable to the evaluation of radiological consequences on which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, 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 NRC staff also considered relevant information in Chapter 14 of the Turkey Point UFSAR, which describes the design-basis accidents (DBA) and evaluation of their radiological consequences.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition." SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0 (ADAMS Accession No. ML003734190), provides guidance to the NRC staff for the review of AST amendment requests. SRP Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183.

NUREG-1431, Revision 4, "Standard Technical Specifications [STS] – Westinghouse Plants" (ADAMS Accession No. ML12100A222) contains the STS for Westinghouse plants. Although the Turkey Point 3 and 4 TSs are not based on the guidance in NUREG-1431, the STS present an acceptable method for licensees of Westinghouse plants to meet the NRC's requirements in 10 CFR 50.36.

RG 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (ADAMS Accession No. ML003740282), describes a method acceptable to the NRC staff for complying with the NRC's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant.

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792) provides the methodology for analyzing the radiological consequences of several design basis accidents to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of AST (also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and plant-specific design and licensing basis information discussed in Section 2.3 of this safety evaluation.

3.1 Evaluation of Changes to TS 3/4.3.2

3.1.1 Revision of ACTION 16 in Table 3.3-2

ACTION 16 currently directs compliance with the ACTION requirements of FU 1a of Table 3.3-4 (i.e., ACTION 26) in the event that an applicable FU does not meet its minimum channel operable requirement. ACTION 16 currently applies to FUs 3.c.2, 3.c.4, 9.a, and 9.c. The proposed change would revise ACTION 16 to require the containment purge, exhaust, and instrument air bleed valves to be maintained closed in the event that an applicable FU does not meet its minimum channel operable requirement, while allowing the instrument air bleed valves to be opened intermittently under administrative controls. The revised ACTION 16 would only be applicable to FUs 3.c.2 and 3.c.4. The relevant changes to FUs 9.a and 9.c are evaluated in Sections 3.1.3 and 3.1.6 below.

As it relates to FU 3.c.2, the current ACTION 16 could allow indefinite operation with less than the minimum automatic actuation logic and actuation relay channels without requiring any remedial actions to be taken, if either the Particulate (R-11) or Gaseous (R-12), Containment Radioactivity monitors are operable. This is because ACTION 26 is only applicable if both R-11 and R-12 are inoperable. As such, the existing ACTION 16 is non-conservative. The licensee stated in its application that it has administrative controls in place that require the containment purge, exhaust, and instrument air bleed valves to be maintained closed whenever the minimum channels operable requirement of FU 3.c.2 is not met.

The proposed ACTION 16 resolves the non-conservatism with FU 3.c.2 by requiring the containment purge, exhaust, and instrument air bleed valves to be maintained closed in the event that FU 3.c.2 does not meet its minimum channel operable requirement, while still allowing the instrument air bleed valves to be opened intermittently under administrative controls. This is consistent with the relevant remedial action required by ACTION 26. Similarly, the proposed ACTION 16 maintains the requirement of ACTION 26 to close the containment purge, exhaust, and instrument air bleed valves in the event that FU 3.c.4 does not meet its minimum channel operable requirement (i.e., both R-11 and R-12 are inoperable), without requiring remedial actions associated with the RCS leakage detection system in the existing ACTION 26. The NRC staff's evaluation of the licensee's proposed change to ACTION 26 is in Section 3.2.4, below.

For the reasons discussed above, the NRC staff finds that the proposed ACTION 16 resolves the non-conservatism associated with FU 3.c.2 while maintaining the existing remedial actions required for FU 3.c.4. The proposed ACTION 16 does not change the administration of or actions associated with ESFAS instrumentation. Therefore, the proposed change does not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and does not alter the manner in which the subject instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. Further, the NRC staff finds this proposed change to be consistent with the DBAs radiological consequences assumptions and has determined that there is no impact to the calculated radiological doses. As a result, the NRC staff finds the proposed change acceptable.

3.1.2 Revision of Notation ## in Table 3.3-2 and Deletion of Notation (1) in Table 3.3-3

Notation (1) in Table 3.3-3 states that, "[e]ither the particulate or gaseous channel in the OPERABLE status will satisfy this LCO." Notation (1) currently applies to both FU 3.c.4 and FU 9.c. The NRC staff's evaluation of the licensee's proposed changes to FU 9.c is in Section 3.1.6, below.

The licensee proposes to delete Notation (1) in Table 3.3-3 and modify Notation ## in Table 3.3-2 by adding the following text:

> Either an OPERABLE particulate radioactivity or gaseous activity channel will satisfy the Minimum Channels OPERABLE requirement.

The revised Notation ## would apply to FU 3.c.4.

The NRC staff notes that in Table 3.3-2, the Total Number of Channels requirement of FU 3.c.4 is two, and the Minimum Channel Operable requirement is one. Operability of either the Particulate (R-11) or the Gaseous (R-12) radioactivity channel would satisfy the Minimum Channel Operable requirement. Therefore, the proposed Note ##, as shown above, is consistent with the Minimum Channel Operable requirement of FU 3.c.4 and the intent of Note (1).

The NRC staff finds that proposed Note ## would continue to be an applicable condition for FU 3.c.4 and does not change the administration of or actions associated with ESFAS instrumentation. Also, the NRC staff finds that with the proposed Note ##, Note (1) in Table 3.3-3 is no longer necessary. As such, the NRC staff finds that the deletion of Note (1) is acceptable. Therefore, the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and do not alter the manner in which the subject instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. Further, the NRC staff finds the proposed changes to be consistent with the DBAs radiological consequences assumptions and has determined that there is no impact to the calculated radiological doses. As a result, the NRC staff finds the proposed changes acceptable.

3.1.3 Revision of ACTION 24 and Application to FU 9.a and FU 9.e in Table 3.3-2

Currently, Table 3.3-2 invokes ACTION 16 when the number of operable channels for FU 9.a is less than the minimum number of operable channels (2). For reasons similar to discussed in Section 3.1.1, above, for FU 3.c.2, ACTION 16 could allow indefinite operation with less than the minimum automatic actuation logic and actuation relay channels without requiring any remedial actions to be taken, if either the Particulate (R-11) or Gaseous (R-12), Containment Radioactivity monitors are operable. As such, the existing ACTION 16 is non-conservative. To address this non-conservatism, the licensee proposes to revise ACTION 24, redesignate the revised statement as ACTION 24A and create new ACTION 24B.

ACTION 24 currently applies to FU 9.e and requires CREVS to be isolated and placed in recirculation mode within 1 hour when the number of operable channels is one less than the minimum number of operable channels. Proposed ACTION 24A would allow a 7-day completion time to restore the inoperable channel to operable status prior to placing the CREVS into recirculation mode.

Proposed ACTION 24B would address the condition in which both control room air intake radiation level channels are inoperable and would require the following:

1. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with BOTH Control Room emergency recirculation fans operating, OR

2. a. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with ONE Control Room emergency recirculation fan operating, AND

b. Restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both Units simultaneously, then be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours.

The proposed seven-day completion time to restore an operable channel for ACTIONS 24A and 24B is consistent with the completion time specified in TS 3.7.5 to restore an inoperable but redundant CREVS component before a plant shutdown is required. Isolating the normal mode of CREVS and operating the CREVS in the recirculation mode accomplishes the function performed by the control room air intake radiation level instrumentation. Further, the proposed seven-day completion time is consistent with STS 3.3.7A, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation (Without Setpoint Control Program)."

Proposed ACTION 24B requires the normal mode of CREVS to be immediately isolated and the recirculation mode to be established. Aligning the CREVS in this configuration accomplishes the function performed by the control room air intake radiation level instrumentation. By requiring both recirculation fans in service, proposed ACTION 24B will ensure that the CREVS function is performed in the presence of any single active failure given the redundancy of the other active CREVS components.

For the reasons discussed above, the NRC staff finds that the proposed ACTIONS 24A and 24B resolve the non-conservatism associated with FU 9.a and are appropriate remedial actions in the event of one or two inoperable channels, respectively, for FU 9.a and FU 9.e. The proposed ACTIONS 24A and 24B do not change the administration of or actions associated with ESFAS instrumentation. Therefore, the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and do not alter the manner in which the subject instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. Further, the NRC staff finds the proposed changes to be consistent with the DBAs radiological consequence assumptions and has determined that there is no impact to the calculated radiological doses, because the CREVS is capable of performing its mitigation functions as assumed in Turkey Point 3 and 4's design basis. As a result, the NRC staff finds the proposed changes acceptable.

The addition of ACTION 24B requires the creation of a new TS page. The new page will be designated 3/4 3-22A. The NRC staff finds the change administrative in nature and acceptable.

3.1.4 Deletion of MODE 6 from List of Applicable Modes and CORE ALTERATIONS from Notation ** for FU 9.a in Table 3.3-2

Currently, FU 9.a of TS Table 3.3-2 is required to be operable in MODES 1, 2, 3, and 4, and in MODE 6 during core alterations or movement of irradiated fuel within the containment. The proposed change deletes MODE 6 and the reference to core alterations in the note denoted by ** in TS Table 3.3-2, such that requirements for FU 9.a would be applicable in MODES 1, 2, 3, and 4, and during the movement of irradiated fuel within the containment. The licensee's application states:

> The bases for the removal was that the accidents postulated to occur during CORE ALTERATIONS do not result in fuel cladding integrity damage. In Reference 6.6 [ADAMS Accession No. ML13246A358], the NRC expressed concern about the dose consequences that could result from the application of [Technical Specifications Task Force] TSTF-51. Consistent with the TSTF Committee recommendations specified in Reference 6.7 [ADAMS Accession No. ML15034A172], FPL has confirmed that the onsite and offsite doses resulting from the unlikely dropping of a load allowed to be moved during CORE ALTERATIONS onto irradiated fuel assemblies seated in the reactor vessel or fuel storage pool are bounded by [the] Turkey Point fuel handling accident (FHA) analysis of record when crediting only safety systems required to be OPERABLE. As such, the proposed change is consistent with the NRC endorsed TSTF-51 and NUREG-1431, Revision 4, Volume 1, Specifications (Reference 6.2) [ADAMS Accession No. ML12100A222], is not applicable to the Staff's recent concern regarding TSTF-51, and is thereby reasonable.

The NRC staff has reviewed the licensee's request to delete core alterations from the TS applicability against NUREG-1431, TSTF-51, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations" (ADAMS Accession No. ML993190284), and the many documents between the NRC staff and the TSTF related to TSTF-51. The NRC staff has determined that because the FHA analysis of record resultant radiological dose for Turkey Point bounds the dropping of light loads onto irradiated fuel assemblies seated in the reactor vessel or fuel storage pool, this proposed change has no impact on the DBAs radiological consequences assumptions, nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective. Therefore, deletion of CORE ALTERATIONS from Notation ** is acceptable.

The NRC staff finds that once the reference core alterations is removed, the MODE 6, "Refueling," applicability is no longer necessary for FU 9.a. This is because the MODE 6 applicability is in place for consistency with the core alterations notation. In addition, the proposed TS applicability requirements for FU 9.a in TS Table 3.3-2 are consistent with and have no impact on the DBAs radiological consequences analyses. As such, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective.

Based on the above, the NRC staff concludes that the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and do not alter the manner in which the subject monitoring instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. As a result, the NRC staff finds the proposed changes acceptable.

3.1.5 Deletion of CORE ALTERATIONS from Notation (4) and Addition to List of Applicable Modes for FU 9.a in Table 4.3-2

The licensee proposes to delete the core alterations reference in Notation (4) of Table 4.3-2 and add to the column, "MODES FOR WHICH SURVEILLANCE IS REQUIRED," in Table 4.3-2, FU 9.a. Currently, TS Table 4.3-2 does not state when to perform the SRs for FU 9.a. The proposed change corrects the inadvertent omission by aligning the applicability for the SRs to be consistent with operability requirements for FU 9.a in TS Table 3.3-2. In addition, the proposed change deletes the reference to core alterations in the note denoted by (4) in TS Table 4.3-2, such that the applicability for FU 9.a SRs will be MODES 1, 2, 3, and 4, and during movement of irradiated fuel within the containment.

The NRC staff has determined that this proposed change provides consistency between the operability requirements and the SRs applicability for FU 9.a and that these requirements are consistent with and have no impact on the DBAs radiological consequence analyses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective. Similarly, and as discussed in Section 3.1.4, the NRC staff finds the removal of core alterations to be acceptable from a radiological dose perspective.

Based on the above, the NRC staff concludes that the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2). As a result, the NRC staff finds the proposed changes acceptable.

3.1.6 Deletion of FU 9.c from Tables 3.3-2, 3.3-3, and 4.3-2

The licensee proposes to delete FU 9.c from TS Tables 3.3-2, 3.3-3 and 4.3-2. Currently, if a high containment radioactivity signal is generated from FU 9.c, the control room ventilation isolation signal will occur, which isolates the normal mode of CREVS and establishes the recirculation mode. However, in the event there are less than the minimum number of operable channels for FU 9.c, Table 3.3-2 invokes ACTION 16, which directs the operator to ACTION 26 in Table 3.3-4. ACTION 26 requires remedial actions related to containment isolation and RCS leak testing, but does not address isolating the control room. Therefore, the existing ACTION 16 is non-conservative. The licensee stated in its application that it has administrative controls in place that require isolation of control room ventilation and placement of the CREVS in recirculation mode whenever the minimum operable channel requirement of FU 9.c. is not met. The licensee stated that placing the CREVS in the emergency recirculation mode whenever the containment atmosphere radioactivity monitoring instrumentation are removed from service accrues CREVS filtration hours towards the 720 hour operability limit imposed by TS 3.7.5, and that routine maintenance has the unintended burden of requiring more frequent control room air in leakage and emergency filtration testing, given the time required to complete these activities.

The licensee proposes to delete FU 9.c to alleviate the accrual of CREVS charcoal and high-efficiency particulate air-filtration hours each time the containment particulate or gaseous radioactivity monitors are removed from service.

The radiological consequences for the following DBAs were approved by the NRC staff on June 23, 2011 (ADAMS Accession No. ML110800666), and are reflected in the Turkey Point 3 and 4 UFSAR, Section 14:

- Loss of Coolant Accident (LOCA)
- FHA
- Main Steam Line Break Accident
- Steam Generator Tube Rupture Accident
- Reactor Coolant Pump Shaft Seizure Accident (Locked Rotor)
- Rod Cluster Control Assembly (RCCA) Ejection Accident
- Waste Gas Decay Tank Rupture

The NRC staff's review of radiological consequences for the DBAs determined the LOCA, FHA and RCCA ejection accidents credit FU 9.c to isolate the normal mode of CREVS and establish the recirculation mode. For the LOCA analysis, the CREVS is initially assumed to be operating in normal mode. After the start of the event, CREVS transitions from normal mode to recirculation mode based on a signal from either FU 9.b, "Safety Injection," or a high radiation signal from either FU 9.c or FU 9.e. For the FHA in containment and the RCCA ejection accident release to containment analysis, the CREVS is initially assumed to be operating in normal mode and transitions to recirculation mode based on a high radiation signal from FU 9.c. Because the proposed change affects the assumptions in these radiological consequence analyses the licensee is proposing to revise the radiological consequences so that they credit FU 9.e (control room air intake radiation monitors), for the control room ventilation isolation function.

In the amendment request, the licensee states that the revised analyses support the assertion that the containment radioactivity monitoring instrumentation need not be credited for the control room ventilation isolation function for any DBA and that they performed supplemental analyses that postulated radiological releases of insufficient dosage at the control room air intake monitors to trigger automatic control room isolation. The licensee stated that the results of the supplemental analyses also demonstrate that with less than the limiting release and a 30-minute delayed control room operator action, the resulting control room doses remain within the regulatory limits.

In Attachment 3 of the amendment request, "Numerical Applications NAI-1983-001, Revised Turkey Point AST Dose Assessment without Credit Containment Radiation Monitors, Revision 1," the licensee states that the changes addressed in the report are limited to the changes to the licensing basis radiological analyses resulting specifically from the elimination of credit assumed for the use of the containment radiation monitors in the current radiological analysis of record as reflected in the UFSAR and that the following remain unchanged:

- Compliance with regulatory guides
- Radiological evaluation methodology
- Control room ventilation system performance parameters
- Control room dose calculation model – RADionuclide Transport, Removal and Dose Estimation analytical code (RADTRAD) NAI
- Radiation source terms for the events
- Atmospheric dispersion factors for onsite and offsite
- Direct shine dose is consistent with the analysis of record, the LOCA shine dose contribution is bounding for all other events

3.1.6.1 Loss-of Coolant Accident

The licensee determined that the LOCA radiological consequence analysis did not need to be reanalyzed because the safety injection signal will provide a containment ventilation and control room isolation signal within the same time as that assumed in the current analysis of record as reflected in Turkey Point's UFSAR; therefore, there is no impact on the LOCA radiological consequences. However, it was not clear to the NRC staff why the timing is the same given the differences in the instrumentation. Therefore, the NRC staff asked the licensee to explain why the timing for the control room isolation signal remains 30 seconds while accounting for the time to reach the signal, the diesel generator start time, load sequencing and damper actuation and positioning time, given the differences between the safety injection instrumentation and the containment radioactivity instrumentation.

In letter dated June 12, 2018, the licensee provided the timing sequences associated with the low pressurizer pressure safety injection signal. These sequences show that the timing of the control room isolation and the CREVS actuation in the current analysis of record will remain the same. Since the low pressurizer pressure safety injection signal remains the credited signal for automatic control room isolation and CREVS actuation and the licensee is not requesting credit of the control room normal air intake radiation monitors in their LOCA radiological consequence analysis, the NRC staff has determined that the proposed change has no impact on the LOCA radiological consequence analysis. Therefore, the NRC staff finds the proposed change to be acceptable and does not require reanalyzing the LOCA radiological consequence analysis.

3.1.6.2 FHA in Containment

Section 14.2.1.2 of the UFSAR describes the FHA as a drop of a single fuel assembly either inside of containment or in the fuel-handling building. The FHA assumes all of the fuel rods in a single fuel assembly are damaged and considers a drop inside the containment with the equipment hatch open, and a drop inside the fuel-handling building without credit for filtration of the fuel-handling building exhaust. However, only the containment release case is impacted by the proposed change to delete FU 9.c, since it assumes that the control room isolation and transition to recirculation mode occurs on a high radiation signal from the containment radiation monitors. The licensee proposes to revise the FHA in containment analysis to remove the containment radiation monitors and to credit the control room normal intake radiation monitors to perform the automatic control room isolation function.

In order to evaluate the control room habitability for the postulated design basis FHA, the licensee analyzed two cases for the FHA in containment. The first case assumes the CREVS is operating in normal mode. The assumed airflow during the normal mode of operation is 1000 cubic feet per minute (cfm) of unfiltered fresh air makeup and an unfiltered in leakage of 100 cfm. After the start of the event, the control room will be isolated on a high radiation signal from the control room normal intake radiation monitor. The licensee applied a 60-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 control room isolation, the airflow 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 in leakage, and 375 cfm of filtered recirculation flow. The current licensing basis assumes a 30-second delay time. FPL has not proposed any other changes to the current licensing basis source term, inputs, assumptions or methodology for the FHA in containment. The licensee's analysis determined that the control room radiological dose increased from 1.22 rem to 1.44 rem, which remains below the control room dose regulatory limit. The licensee also determined that the radiological consequences at the exclusion area boundary (EAB), and the low population zone (LPZ) remain unchanged from the current licensing basis since the proposed change only affects the control room isolation time.

For the second case, FPL performed a supplemental analysis to investigate a postulated reduced radiological release that would not produce a sufficient radiological dose at the control room normal intake radiation monitor to trigger an automatic control room isolation. The control room ventilation system is initially assumed to be operating in normal mode. The airflow assumed during the normal mode of operation is 1000 cfm of unfiltered fresh air makeup and an unfiltered in leakage of 100 cfm. The control room is assumed to be manually isolated by operator action 30 minutes after the initiating event. After control room isolation, the airflow 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 in leakage, and 375 cfm of filtered recirculation flow. This case assumes that the source term is reduced to 8 percent of the current licensing basis source term, which does not actuate the automatic control room isolation and transition to recirculation mode. Other than the changes discussed above, FPL has not proposed any other changes to the current licensing basis inputs, assumptions or methodology for the FHA in containment. The licensee's analysis determined that the control room radiological dose is 1.75 rem, which remains below the regulatory control room dose limit.

In accordance with SRP 15.0.1 Section III.6.c, to verify the licensee's analyses, the NRC staff performed confirmatory radiological consequence dose calculations and compared the results to those calculated by the licensee. However, the NRC staff's resultant radiological doses differed from the licensee's results, therefore to resolve the differences the NRC staff performed an audit of the following document:

- NAI-1396-012, Revision 4, "Turkey Point EPU [Extended Power Uprate] Fuel Handling Accident Radiological AST Analysis with High Burnup Fuel," dated June 12, 2017.

Based on the review of the document above, the NRC staff was able to verify the licensee's analyses and confirm their resultant radiological doses. The NRC staff performed independent confirmatory dose calculations to ensure a thorough understanding of the licensee's methods. The radiological consequences calculated by the NRC staff for the EAB, LPZ, and control room are consistent with those calculated by the licensee. The NRC staff finds that the radiological consequences at the EAB and LPZ remain unchanged from the current licensing basis and control room doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are, therefore acceptable.

3.1.6.3 RCCA Ejection Accident Release to Containment

Section 14.2.6 of the UFSAR describes the RCCA ejection accident as the mechanical failure of a RCCA and drive shaft resulting in the ejection of the 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.

One case assumes that all of the radioactivity released from the damaged fuel is fully dispersed in the primary coolant system and subsequently released to the secondary system through steam generator tube leakage and to the environment via steaming from the atmospheric dump valves. The secondary release analysis of record relies on the control room normal intake radiation monitor to accomplish the CREVS transition to recirculation mode. This case also investigated a postulated reduced radiological release that would not produce a sufficient radiological dose at the control room normal intake radiation monitor to trigger an automatic control room isolation. Instead, it relies upon operators manually isolating the control room in 30 minutes. Because neither the automatic nor manual control room isolation case credits the containment radiation monitor, the secondary release cases are not impacted by the proposed change, therefore, the NRC staff finds the proposed change to be acceptable from a radiological dose perspective.

The second 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 licensing basis credits CREVS transition to recirculation mode based upon a high radiation signal from the containment radiation monitors. Therefore, the licensee revised the analysis to remove the reliance upon the containment radiation monitors and to credit the control room normal intake radiation monitors to perform the automatic control room isolation function as was done in the secondary release cases.

Attachment 3, Section 2.1.2, "RCCA Ejection – Containment Release Event Analysis," of the license amendment request states:

> For the new Containment Release cases, the Control Room is isolated on a high radiation reading at the CR [control room] normal intake monitors. A 60 second delay is applied to account for the time to reach the setpoint (30 seconds), signal processing, and damper closure time for the automatic CR isolation case. Previously the isolation conservatively assumed 60 seconds.

However, it was not clear to the NRC staff why the timing is the same given the differences in the instrumentation. Therefore, the NRC staff asked the licensee to explain why the timing for the control room isolation signal remains 60 seconds while accounting for the time to reach the setpoint, signal processing, and damper closure time, given the differences between the control room normal intake instrumentation and the containment radioactivity instrumentation.

In the letter dated June 12, 2018, the licensee acknowledged a discrepancy between UFSAR Section 14.2.6.4 and UFSAR Table 14.2.6-3 regarding the credited time for isolating the normal control room intake ventilation and placing CREVS in service and stated that UFSAR Section 14.2.6.4 is correct and consistent with the RCCA ejection radiological dose consequence analysis for both the containment and that the secondary release cases credit

60 seconds. In addition, the licensee provided the timing sequences associated with the control room radiation monitor signal, which shows that the timing of the control room isolation and the CREVS actuation is bounded by the current analysis of record. Since the control room intake radiation monitors were shown to automatically isolate the control room within the same time assumed in the current containment release case for the RCCA ejection analysis of record, and because the licensee does not propose any other changes to the analysis of record for the automatic isolation case, the resultant radiological doses remain unchanged. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective.

The secondary release case, in the current licensing basis, states:

> The secondary release scenario credits control room isolation from a high radiation signal on the control room intake monitor. The Technical Specification setpoint for this instrument is 2 mR/hr [milliroentgen per hour]. In the RCCA Ejection analysis, an analytical setpoint of 5 mR/hr was used to account for measurement and test uncertainties and to apply additional conservatism. For the design basis fuel failure and core melt fractions, the calculated exposure rate at the detector exceeded the analytical setpoint by approximately 35%. It was recognized that with only 35% margin, a scenario could be postulated with fuel failure fractions less than the design values in which the analytical setpoint would not be reached and a delayed manual isolation must be assumed. While the offsite dose consequences would be lower in such a scenario, the relative impact of lower fuel failure fractions with a longer control room isolation time was not immediately obvious. Therefore, an additional case was performed which combined the reduced source term with a 30-minute control room isolation time.

To be consistent with the methodology stated in the above licensing basis, the licensee performed a supplemental analysis to investigate the impact of manually isolating the control room at 30 minutes for a reduced radiological release that does not produce the control room isolation signal at the control room normal intake radiation monitor. The inputs and assumptions of the supplemental analysis are the same as the current licensing basis, as reflected in UFSAR Section 14.2.6, with the exception of the following:

- The source term is reduced from 10 percent breached fuel to 2.31 percent and from 0.25 percent clad melt to 0.05775 percent; and
- Control room isolation occurs manually at 30 minutes instead of automatically at 1 minute.

In accordance with SRP 15.0.1 Section III.6.c, to verify the licensee's analyses, the NRC staff performed confirmatory radiological consequence dose calculations and compared the results to those calculated by the licensee. However, the NRC staff's resultant radiological doses differed from the licensee's results, therefore to resolve the differences the NRC staff performed an audit of the following document:

- NAI-1396-027, Revision 4, "Turkey Point EPU RCCA Ejection Radiological Analysis with Alternative Source Term," dated June 12, 2017.

Based on the review of the document above, the NRC staff was able to verify the licensee's analyses and confirm their resultant radiological doses. The NRC staff performed independent confirmatory dose calculations to ensure a thorough understanding of the licensee's methods. The radiological consequences calculated by the NRC staff for the EAB, LPZ, and control room are consistent with those calculated by the licensee. The NRC staff finds that the radiological consequences at the EAB and LPZ remain unchanged from the current licensing basis and control room doses estimated by the licensee for the RCCA ejection accident release to containment meets the applicable accident dose criteria and is, therefore acceptable.

3.1.6.4 Conclusion

The NRC staff's review of the RCCA ejection accident release to containment and the FHA in containment has found that the licensee used analysis assumptions and inputs consistent with SRP Section 15.0.1 and RG 1.183. The licensee's calculated dose results are given in Table 1, below, and the assumptions found acceptable to the NRC staff are presented in Tables 2 and 3, below. To verify the licensee's analyses, the NRC staff performed confirmatory radiological consequence dose calculations and compared the results to those calculated by the licensee. The radiological consequences calculated by the NRC staff for the control room are consistent with those calculated by the licensee. Moreover, the radiological consequences calculated by both the licensee and the NRC staff are well within the radiation dose criteria set forth in 10 CFR 50.67. The NRC staff finds that the control room doses estimated by the licensee for both accidents meet the applicable accident dose criteria and that the EAB and LPZ doses are bounded by the current licensing basis, and are therefore acceptable. The NRC staff has determined that the proposed change to delete FU 9.c, from TS Tables 3.3-2, 3.3-3 and 4.3-2 is acceptable from a radiological dose perspective.

Based on the above, the NRC staff concludes that FU 9.c no longer meets the criteria of 10 CFR 50.36(c)(2)(ii) and is therefore no longer required to be retained as an LCO in the TSs per the regulations at 10 CFR 50.36(c)(2). As a result, the NRC staff finds the proposed deletion of FU 9.c from TS Tables 3.3-2, 3.3-3 and 4.3-2 acceptable.

Table 1. Control Room Radiological Dose Consequences

	Current Control Room Dose in rem	Proposed Control Room Dose in rem
Automatic Isolation FHA in containment	1.22	1.44
Manual Isolation RCCA Ejection in containment FHA in containment	Not Applicable Not Applicable	4.43 1.75

Table 2. FHA Inputs and Assumptions

Input/Assumption	**Value**
Core Power Level before shut down	2652 megawatts thermal
Discharged Fuel Assembly Burnup	45,000 megawatt days per metric ton of uranium
Fuel Enrichment	3.0 to 5.0 w/o
Radial Peaking Factor	1.65
Number of Fuel Assemblies Damaged Automatic Control Room Isolation Manual Control Room Isolation	1 assembly 0.08 assembly
Release Fraction from Breached Fuel	UFSAR Table 14.2.1-1
Delay before Spent Fuel Movement	72 hours

Input/Assumption	Value
Release Duration	2 hours
FHA Source Term for a Single Assembly	UFSAR Table 14.2.1-2
Water Level above damaged fuel assembly	23 feet minimum
Iodine Decontamination Factors	Elemental 285 Organic 1
Atmospheric Dispersion Factors Offsite Onsite	 UFSAR Appendix 2E UFSAR Appendix 2F
Control Room Ventilation System Containment release Auto isolation on control room intake radiation monitor Manual isolation with reduced source term to 8% of design release	 60 seconds 30 minutes
Control Room Unfiltered in leakage	100 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 3. RCCA Ejection Accident Release in Containment Inputs and Assumptions

Input/Assumptions	Value
Core Power Level before shut down	2652 megawatts thermal
Discharged Fuel Assembly Burnup	45,000 megawatt days per metric ton of uranium
Fuel Enrichment	3.0 to 5.0 w/o
Radial Peaking Factor	1.65
Percent of Core in Departure from Nucleate Boiling Design Basis Scenario Manual Control Room Isolation	 10% 2.31%
Percent of Core with Centerline Melt Design Basis Scenario Manual Control Room Isolation	 0.25% 0.05775%
Gap Release Fraction	RG 1.183 Appendix G Position 1
Core Fission Product Inventory	UFSAR Table 14.3.5-7
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm dose equivalent I-131
Release from Departure from Nucleate Boiling	RG 1.183 Appendix H section 1
Release from Fuel Centerline Melt Fuel	RG 1.183 Appendix H section 1
Chemical Form of Iodine Released to Containment	Particulate 95% Elemental 4.85% Organic 0.15%
Atmospheric Dispersion Factors Offsite Onsite	 UFSAR Appendix 2E UFSAR Appendix 2F
Control Room Ventilation System Time of Automatic Control Room Isolation Time of Manual Control Room Isolation	 60 seconds 30 minutes

Input/Assumptions	Value
Control Room Unfiltered in Leakage	100 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6
Containment Volume	1.60E+06 cubic feet
Containment Leakage Rate 0 to 24 hours After 24 hours	 0.20% by weight per day 0.10% by weight per day
Containment Natural Deposition Coefficients	Aerosols 0.1 /hour Elemental Iodine 5.58 hour Organic Iodine None

3.2 Evaluation of Changes to TS 3/4.3.3.1

3.2.1 Deletion of FU 1.b from Tables 3.3-4 and 4.3-3

The licensee proposes to delete the RCS leakage detection particulate and gaseous radioactivity instrumentation, FU 1.b, from TS Tables 3.3-4 and 4.3-3. FU 1.b requires the RCS leakage detection particulate radioactivity or gaseous radioactivity instrumentation to be operable while in MODES 1 through 4. TS LCO 3.4.6.1 requires the same RCS leakage detection particulate radioactivity or gaseous radioactivity instrumentation for the RCS leakage detection function in MODES 1 through 4. ACTION 26 in Table 3.3-4 and TS LCO 3.4.6.1, ACTION a are both entered into when both RCS leakage detection particulate radioactivity and gaseous radioactivity instrumentation channels are inoperable and both require the same remedial actions. Table 4.3-3 requires a channel check, channel calibration, and analog channel operational test to be performed while in MODES 1 through 4 in accordance with the Turkey Point Surveillance Frequency Control Program (SFCP). SR 4.4.6.1 also requires a channel check, channel calibration, and analog channel operational test to be performed while in MODES 1 through 4, but refers to TS Table 4.3-3, for the SR frequencies. As part of this change, the licensee has proposed to replace the reference in SR 4.4.6.1 to Table 4.3-3 with a reference to the Turkey Point SFCP. The NRC staff's evaluation of that proposed change is discussed in Section 3.3.2, below.

The NRC staff finds that the proposed change to delete FU 1.b from Tables 3.3-4 and 4.3-3 is administrative in nature because the LCO, ACTION, and SR associated with RCS leakage detection remain the same. As such, the proposed change is acceptable.

3.2.2 Removal of CORE ALTERATIONS from Notation * in Table3.3-4

Notation * in Table 3.3-4 is associated with FU 1.a and states that during core alterations or movement of irradiated fuel within the containment compliance with TS 3/4.9.13, "Radiation Monitoring," is required. The proposed change deletes the reference to core alterations in Notation * in Table 3.3-4. In addition, the proposed change adds a comma between the words "containment" and "comply" for grammatical correctness in the note. The NRC staff finds the addition of a comma to be administrative in nature and, therefore, acceptable.

As discussed in Section 3.1.4, the NRC staff has determined that because the FHA analysis of record resultant radiological dose for Turkey Point bounds the dropping of light loads onto irradiated fuel assemblies seated in the reactor vessel or fuel storage pool, that the deletion of core alterations in Notation * of TS Table 3.3-4 has no impact on the DBAs radiological consequences assumptions nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective.

Based on the above, the NRC staff concludes that the proposed change does not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and does not alter the manner in which the subject monitoring instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. As a result, the NRC staff finds the proposed change acceptable.

3.2.3 Relocation of the Requirements of FUs 2.a and 2.b from Table 3.3-4 and Table 4.3-3 to Licensee-Controlled Documents and Deletion of the Associated Notations and ACTION 28

The licensee proposes to relocate FUs 2.a and 2.b from Tables 3.3-4 and 4.3-3 to the Turkey Point 3 and 4 ODCM and applicable plant procedures, where future changes to the relocated requirements would be subject to the requirements of 10 CFR 50.59. The proposed change also deletes ACTION 28 in Table 3.3-4, and the Notations associated with FUs 2.a and 2.b—Notations # and ** in Table 3.3-4 and Notation * in Table 4.3-3.

In the Turkey Point 3 and 4 UFSAR, Section 11.2.3, "Radiation Monitoring System," identifies the affected instruments that comprise FUs 2.a and 2.b as:

- (Unit 3 only) spent fuel pool (SFP) exhaust monitors: RAD-3-6418, "SFP Vent Exhaust SPING Monitor - Unit 3" and RD-3-1419, "Spent Fuel Pit Exhaust."

- Plant Vent Exhaust radioactivity monitors: RAD-6304, "Vent Stack Wide Range Monitors" and R-14, "Plant Vent Gas Monitors."

The NRC staff used the criteria of RG 1.97 to verify that the instruments of FU 2.a and FU 2.b are neither Type A nor Category 1. The Reviewer's Note found in NUREG-1431, STS 3.3.3, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," states: "Table 3.3.3-1 shall be amended for each unit as necessary to list: (1) All Regulatory Guide 1.97, Type A instruments and (2) All Regulatory Guide 1.97, Category 1, non-Type A instruments in accordance with the unit's Regulatory Guide 1.97, Safety Evaluation Report." Therefore, if the instrumentation associated with FUs 2.a or 2.b are either Type A instrumentation or Category 1, according to Table 3, "PWR [pressurized-water reactor] Variables," of RG 1.97, then those FUs would need to be retained in the TSs.

The following table compares the instrumentation associated with FU 2.a and FU 2.b with its corresponding RG 1.97 type and category:

Table 4. RG 1.97 Type and Category Information for FU 2.a and FU 2.b Instrumentation

Instrument Number	Variable Description	RG 1.97 Type	RG 1.97 Category
RAD-3-6418 (Unit 3 only)	SFP Vent Exhaust SPING Monitor (containment effluent radioactivity noble gas)	C	2
	SFP Vent Exhaust SPING Monitor (all other identified release points)	E	2

	SFP Vent Exhaust SPING Monitor (particulates and halogens)	E	3
RD-3-1419 (Unit 3 only)	Area Radiation Monitor (SFP area radiation)	E	3
RAD-6304	Plant Vent Stack Monitor (containment effluent noble gas from buildings or area and other identified release points)	C	2
	Plant Vent Stack Monitor (common vent noble gas)	E	2
	Plant Vent Stack Monitor (particulates and halogens)	E	3
R-14	Plant Vent Exhaust Area Radiation Monitor	N/A	N/A

The Turkey Point 3 and 4 UFSAR analysis for RG 1.97 instrumentation does not list the monitor R-14. The licensee stated in its request that the monitor RaD-6304 encompasses the range of R-14.

As shown in Table 4 above, the instrumentation associated with FU 2.a and FU 2.b are neither Type A nor Category 1. Therefore, they are not necessary to be included in Table 3.3-4 or Table 4.3-3. Additionally, the NRC staff has determined that FU 2.a and FU 2.b are not credited in any of the design basis accident radiological consequence analyses; therefore, their relocation from TS to the ODCM and the deletion of the associated Notation # and Notation ** in Table 3.3-4 and Notation * in Table 4.3-3 and its associated ACTION 28 in Table 3.3-4 have no impact on the DBAs radiological consequences assumptions, nor does it impact the DBAs calculated radiological doses. As such, FU 2.a and FU 2.b do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion as an LCO.

Based on the above, NRC staff concludes that the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3). As a result, the NRC staff finds the proposed changes acceptable.

3.2.4 Revision of ACTION 26 of Table 3.3-4

ACTION 26 in Table 3.3-4 currently directly applies to FU 1.a during MODES 1 through 4 and FU 1.b and allows operation to continue for 7 days in the event that both the containment particulate and gaseous radioactivity monitors are inoperable, as long as: (1) the containment sump level monitoring system is operable, (2) appropriate grab samples are obtained and analyzed once per 24 hours, (3) a reactor coolant system water inventory balance is performed every 8 hours, and (4) containment purge, exhaust and instrument air bleed valves are closed. ACTION 26 also currently applies to the containment or control room ventilation isolation instrumentation requirements (FU 3.c or FU 9, respectively) of Table 3.3-2, because ACTION 16 directs compliance with ACTION 26. However, ACTION 26 does not address all the above functions impacted by inoperable containment radioactivity monitoring instrumentation, but instead specifies the same required actions as ACTION a of TS 3.4.6.1 for inoperable containment atmosphere gaseous and particulate radioactivity monitoring systems.

The licensee proposes to revise ACTION 26 to state:

> In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, comply with the following:
>
> 1) Table 3.3-2, ACTION 16, and
>
> 2) Technical Specification 3.4.6.1, ACTION a.

The revised ACTION 26 would apply to FU 1.a in Table 3.3-4. The NRC staff's evaluation of the licensee's proposed deletion of FU 1.b is in Section 3.2.1, above. The NRC staff's evaluation of the licensee's proposed changes to ACTION 16 is in Section 3.1.1, above.

The proposed change revises ACTION 26 to address the specific functions impacted by the inoperable containment radioactivity monitoring instrumentation. Specifically, revised ACTION 26 directs compliance with proposed ACTION 16 of Table 3.3-2 for the containment ventilation isolation function, and to existing ACTION a of TS 3.4.6.1 for the containment atmosphere gaseous and particulate radioactivity monitoring systems function. Thus, the proposed change provides actions that address all the functions provided by the containment particulate and gaseous radioactivity monitoring instrumentation.

The NRC staff finds this proposed change is consistent with the DBAs radiological consequence assumptions and has determined that there is no impact to the calculated radiological doses, because the CREVS is capable of performing its mitigation functions as assumed in Turkey Point's design basis. Further, the proposed ACTION 26 does not change the administration of or actions associated with Radiation Monitoring Instrumentation for Plant Operation. Therefore, the proposed change does not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and does not alter the manner in which the subject instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. As a result, the NRC staff finds the proposed change acceptable.

3.2.5 Removal of CORE ALTERATIONS from ACTION 27 in Table 3.3-4

The proposed change deletes the references to core alterations in ACTION 27 of Table 3.3-4, such that the applicability for FU 1.a will be MODES 5 and 6, and during movement of irradiated fuel within the containment.

As discussed in Section 3.1.4, the NRC staff has determined that because the FHA analysis of record resultant radiological dose for Turkey Point bounds the dropping of light loads onto irradiated fuel assemblies seated in the reactor vessel or fuel storage pool that the deletion of core alterations in ACTION 27 of Table 3.3-4 has no impact on the DBAs radiological consequences assumptions nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective.

Based on the above, the NRC staff concludes that the proposed change does not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and does not alter the manner in which the subject monitoring instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. As a result, the NRC staff finds the proposed change acceptable.

3.3 Evaluation of Changes to TS 3/4.4.6

3.3.1 Deletion of TS 3.4.6.1, ACTION a.4

When both the containment particulate and gaseous radiation monitoring systems are inoperable, Action a.4 of TS 3.4.6.1 requires maintaining the containment purge, exhaust and instrument air bleed valves in the closed position. The proposed change deletes ACTION a.4 of TS 3.4.6.1, "Reactor Coolant System Leakage Detection Systems," and the associated Notation **, which permits intermittently opening the instrument air bleed valves under administrative control. The proposed change adds these requirements to the newly proposed ACTION 16 of TS Table 3.3-2. As discussed in Section 3.1.1, proposed ACTION 16 of Table 3.3-2 requires isolation of the containment purge, exhaust and instrument air bleed valves when the minimum channels requirement for Table 3.3-2, FU 3.c.4, is not met. Similar to ACTION a.4 of TS 3.4.6.1, the minimum channels requirement of functional unit 3.c.4 is not met when both the particulate and gaseous radioactivity monitoring systems are inoperable. In essence, this proposed change relocates the TS requirement from the RCS leakage detection system TS to the ESFAS instrumentation TS, which addresses containment isolation. The NRC staff notes that the APPLICABILITY for LCO. 3.4.6.1 is identical to that for FU 3.c.4.

The NRC staff finds that deleting ACTION a.4 is appropriate because the same remedial measures will be appropriately established in the section of the Turkey Point TSs applicable to the containment ventilation isolation function, rather than the RCS leakage detection function. In particular, the same remedial measures are appropriately relocated to proposed ACTION 16 of Table 3.3-2, which specifically addresses the containment ventilation isolation function initiated on Containment Radioactivity – High. This relocation aligns the required Actions with the TS specified function and consequently, reducing Control Room operator burden on implementing the TS. In addition, the proposed change modifies ACTION a.3 of TS 3.4.6.1 by deleting the conjunction "and" and the semicolon and truncating the sentence with a period. These proposed changes are administrative in nature and do not alter the TS requirements; the requirement remains maintaining the containment purge, exhaust and instrument air bleed valves closed in the event the containment particulate and gaseous monitors are inoperable.

Based on the above, the NRC staff concludes that the proposed changes have no impact on the DBAs radiological consequences assumptions, nor do they impact the DBAs' calculated radiological doses. Therefore, the NRC staff finds these proposed changes to be acceptable from a radiological dose perspective. Further, the NRC staff concludes that the proposed changes to LCO 3.4.6.1 do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and do not alter the manner in which the subject monitoring instrumentation are operated and maintained, consistent with 1967 Proposed GDC 15, 16, 17 and 53. As a result, the NRC staff finds the proposed changes acceptable.

3.3.2 Changes to SR 4.4.6.1

The proposed change replaces the requirement to perform surveillance testing at the frequencies specified in TS Table 4.3-3, with the requirement to perform the testing in accordance with the Turkey Point SFCP, and it deletes a hyphen between the words "System" and "performance" in SR 4.4.6.1.a of TS 3.4.6.1.

SR 4.4.6.1.a and TS Table 4.3-3 contain duplicative SRs for the containment atmosphere particulate and gaseous radioactivity monitors in MODES 1, 2, 3, and 4. However, SR 4.4.6.1.a refers to TS Table 4.3-3 for the SR frequencies, while TS Table 4.3-3 requires the SRs to be performed in accordance with the Turkey Point SFCP. The proposed change revises SR 4.4.6.1.a to no longer reference TS Table 4.3-3 and in unity with TS Table 4.3-3, references the Turkey Point SFCP. This proposed change does not alter the technical requirements for the existing TS SRs. In addition, the deletion of the hyphen is editorial in nature and has no impact on the SRs.

Based on the above, the NRC staff concludes that these proposed changes have no impact on the DBAs radiological consequences assumptions nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds these proposed changes to be acceptable from a radiological dose perspective. Further, the NRC staff concludes that the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(3) and do not alter the manner in which the subject monitoring instrumentation are operated and maintained, consistent with 1967 Proposed GDC 16 and 17. As a result, the NRC staff finds the proposed changes acceptable.

3.4 Evaluation of Changes to TS 3/4.9.9

3.4.1 Changes to LCO 3.9.9, APPLICABILITY and SR 4.9.9

TS 3.9.9 currently requires the containment ventilation system to be operable during core alterations or movement of irradiated fuel within the containment. SR 4.9.9 requires the containment ventilation isolation system to be demonstrated operable within 100 hours prior to the start of and in accordance with the SFCP during core alterations by verifying that containment ventilation occurs on a high radiation test signal from each of the containment radiation monitoring instrumentation channels. The proposed change removes the core alterations applicability of TS 3.9.9 and replaces CORE ALTERATIONS with "movement of irradiated fuel inside the containment" in SR 4.9.9.

The NRC staff compared the proposed changes to the STS. STS 3.3.6A, "Containment Purge and Exhaust Isolation Instrumentation (Without Setpoint Control Program)," provides the corollary to Turkey Point TS 3/4.9.9. The NRC staff notes that the term "Core Alterations" does not appear in the STS. The APPLICABILITY for STS 3.3.6A reads "According to Table 3.3.6-1." STS Table 3.3.6-1, "Containment Purge and Exhaust Isolation Instrumentation," Note (a) reads, "During movement of [recently] irradiated fuel assemblies within containment." The APPLICABLE SAFETY ANALYSES for STS 3.3.6A, located in Volume 2, "Bases," of NUREG-1431 (ADAMS Accession No. ML12100A228), reads, in part:

> "[Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days).]"

Turkey Point TS 3/4.9.3, "Decay Time" requires that "[w]ith the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel." The Bases for TS 3/4.9.3 (ADAMS Accession No. ML18115A114) reads, in part:

> The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses, and ensures that the release of fission product radioactivity, subsequent to a fuel handling

accident, results in doses that are well within the values specified in 10 CFR 50.67 and RG 1.183.

This TS is applicable during movement of recently irradiated fuel assemblies within containment. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours.

Therefore, the NRC staff finds that the proposed changes to TS 3/4.9.9 are consistent with STS.

As discussed in Section 3.1.4, the NRC staff has determined that because the FHA analysis of record resultant radiological dose for Turkey Point bounds the dropping of light loads onto irradiated fuel assemblies seated in the reactor vessel or fuel storage pool that the deletion of core alterations in TS 3/4.9.9 has no impact on the DBAs radiological consequences assumptions nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective. Further, the NRC staff finds that the proposed changes are consistent with 1967 Proposed GDC 15, 17 and 53, that do not alter the manner in which the subject Containment Ventilation Isolation System is operated and maintained, and that the proposed changes do not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and (c)(3). As a result, the NRC staff finds the proposed changes acceptable.

3.5 Evaluation of Changes to TS 3/4.9.13

TS 3.9.13 requires the containment radiation monitors that initiate containment and control room ventilation isolation to be operable during core alterations or movement of irradiated fuel within the containment. The proposed change deletes the reference to the control room ventilation isolation function in the limiting conditions for operation, core alterations, and its associated ACTION b in TS 3.9.13. ACTION b of TS 3.9.13 requires isolating the normal mode of CREVS and placing CREVS into recirculation mode within 1 hour when one or both radiation monitors are inoperable.

The licensee provided revised radiological consequence analyses for the LOCA, RCCA ejection accident release to containment, and FHA in containment, as discussed in Section 3.1.6 above, which now credit the control room air intake radiation monitors for the control room ventilation isolation function instead of the containment radiation monitors. Because these analyses no longer credit the containment radiation monitors to perform the control room ventilation isolation function, the deletion of the control room ventilation isolation function from the LCO and its associated ACTION b from TS 3.9.13 has no impact on the DBAs radiological consequences assumptions nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective.

As discussed in Section 3.1.4, the NRC staff has determined that because the FHA analysis of record resultant radiological dose for Turkey Point bounds the dropping of light loads onto irradiated fuel assemblies seated in the reactor vessel or fuel storage pool that the deletion of core alterations in TS 3.9.13 has no impact on the DBAs radiological consequences assumptions nor does it impact the DBAs calculated radiological doses. Therefore, the NRC staff finds this proposed change to be acceptable from a radiological dose perspective.

Based on the above, the NRC staff concludes that the proposed change does not affect the licensee's ability to comply with the regulatory requirements of 10 CFR 50.36(c)(2) and does not alter the manner in which the subject monitoring instrumentation are operated and maintained, consistent with 1967 Proposed GDC 12, 15, 16, and 17. As a result, the NRC staff finds the proposed changes acceptable.

3.6 Technical Conclusion

The NRC staff reviewed the radiological impact of the licensee's proposed changes to the TSs on previously analyzed radiological consequences of the postulated DBAs at Turkey Point 3 and 4. The NRC staff finds that the licensee's proposed changes to the RCCA ejection accident release to containment and the FHA in containment analyses used methodologies, assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.3 above. The NRC staff performed confirmatory radiological consequence dose calculations and compared the results to those calculated by the licensee. The calculated radiological consequences are well within the radiation dose criteria set forth in 10 CFR 50.67 and GDC 19 of Appendix A to 10 CFR Part 50. The NRC staff finds that the control room doses estimated by the licensee for both accidents meet the applicable accident dose criteria and that the EAB and LPZ doses are bounded by the current licensing basis, and are, therefore, acceptable. Further, the NRC staff finds that there is reasonable assurance that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with the criteria stated above in Section 2.3. Therefore, the NRC staff finds that the proposed changes to the Turkey Point 3 and 4 TSs are acceptable with regard to the radiological consequences of postulated design basis accidents.

Further, based on the above evaluation, the NRC staff concludes that the licensee's requested changes comply with the regulatory requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3) and are consistent with 1967 Proposed GDC 12, 15, 16, 17, and 53. As a result, the NRC staff finds the proposed changes acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida Department of Health) on September 11, 2018 (ADAMS Accession No. ML18256A001), of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20 and 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, which was published in the FR on February 27, 2018 (83 FR 8516), that the amendments involve NSHC, and there has been no public comment on such finding. Accordingly, the 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.

6.0 CONCLUSION

The Commission has concluded, based on the aforementioned considerations, 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) there is reasonable assurance that 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.

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Date: November 14, 2018

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATIONS CHANGES TO ADDRESS NON-CONSERVATIVE ACTIONS FOR CONTAINMENT AND CONTROL ROOM VENTILATION FUNCTIONS (EPID L-2017-LLA-0425) DATED NOVEMBER 14, 2018

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