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

March 19, 2018

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
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 THE ELIMINATION OF CERTAIN TECHNICAL SPECIFICATIONS REPORTING REQUIREMENTS (CAC NOS. MF9601 AND MF9602; EPID L-2017-LLA-0213)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 279 to Renewed Facility Operating License (RFOL) No. DPR-31 and Amendment No. 274 to RFOL No. DPR-41 for Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively. The amendments change the Technical Specifications in response to the application from Florida Power & Light Company dated April 9, 2017, as supplemented by letter dated October 4, 2017.

The amendments revise the Technical Specifications to remove certain reporting requirements, to remove the completion time for restoring spent fuel pool water level, to address inoperability of one of the two parallel flow paths in the residual heat removal or safety injection headers for the Emergency Core Cooling Systems, and to make other administrative changes.

A copy of the related safety evaluation is also enclosed. 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. 279 to DPR-31
2. Amendment No. 274 to DPR-41
3. Safety Evaluation

cc w/enclosures: Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 279
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 April 9, 2017, as supplemented by letter dated October 4, 2017, 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. 279, 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

Brian W. Tindell

Brian W. Tindell, Acting 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: March 19, 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. 274
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 April 9, 2017, as supplemented by letter dated October 4, 2017, 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. 274, 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

Brian W. Tindell

Brian W. Tindell, Acting 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: March 19, 2018

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 279 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 274 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
xvi	xvi
3/4 1-4	3/4 1-4
3/4 2-13	3/4 2-13
3/4 3-44	3/4 3-44
3/4 4-27	3/4 4-27
3/4 5-3	3/4 5-3
3/4 5-9	3/4 5-9
3/4 6-7	3/4 6-7
3/4 7-24	3/4 7-24
3/4 8-3	3/4 8-3
3/4 9-11	3/4 9-11
5-1	5-1
6-1	6-1
6-2	6-2
6-3	6-3
6-15	6-15
6-22	6-22

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. 279, 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. 274, 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.

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ADMINISTRATIVE CONTROLS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less positive than or equal to $+5.0 \times 10^{-5}$ Δk/k/°F for all the rods withdrawn, beginning of cycle life (BOL), for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 Δk/k/°F at 100 % RATED THERMAL POWER.

APPLICABILITY: Beginning of cycle life (BOL) - MODES 1 and 2* only**.
End of life (EOL) - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive or equal to the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

* With K_{eff} greater than or equal to 1.

** See Special Test Exceptions Specification 3.10.3.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of -limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio in accordance with the Surveillance Frequency Control Program when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO in accordance with the Surveillance Frequency Control Program.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the QUADRANT POWER TILT RATIO*, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABLITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

* Exception to the 16 detector thimble requirement of monitoring the QUADRANT POWER TILT RATIO is acceptable when performing Specification 4.2.4.2 using two sets of four symmetric thimbles.

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 448 psig, or

b. The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

NOTE: LCO 3.0.4.b is not applicable when entering MODE 4.

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in Modes 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge flow paths aligned to the RCS cold legs,*

b. Two RHR heat exchangers,

c. Two RHR pumps with discharge flow paths aligned to the RCS cold legs,

d. A flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two flow paths capable of taking suction from the containment sump.

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

ACTION:

a. With one of the following components inoperable:

1. RHR heat exchanger,
2. RHR suction flow path from the containment sump,
3. RHR parallel injection flow path, or
4. SI parallel injection flow path

Restore the inoperable component to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. Deleted

c. With one of the four required Safety Injection pumps or its associated discharge flow path inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump or flow path to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

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*Only three Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge flow paths aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specification 4.0.4 are not applicable.

#Inoperability of the required diesel generators does not constitute inoperability of the associated Safety Injection pumps.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, the following ECCS components and flow path shall be OPERABLE:

a. One OPERABLE RHR heat exchanger,

b. One OPERABLE RHR pump, and

c. An OPERABLE flow path capable of (1) taking suction from the refueling water storage tank upon being manually realigned and (2) transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no OPERABLE ECCS flow path from the refueling water storage tank, restore at least one ECCS flow path to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With either the residual heat removal heat exchanger or RHR pump inoperable, restore the components to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

SURVEILLANCE REQUIREMENTS

4.5.3 The above ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between 90% and 95% of the predicted force, or with one tendon below 90% of the predicted force, restore the tendon(s) to the required level of integrity within 15 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the average of all measured tendon forces for each type of tendon (dome, vertical, and hoop), including those measured in ACTION a., less than the predicted force, restore the tendon(s) to the required level of integrity within 15 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any abnormal degradation of the structural integrity other than ACTION a. and ACTION b., at a level below the acceptance criteria of Specifications 4.6.1.6.1, 4.6.1.6.2 and 4.6.1.6.3, restore the containment to the required level of integrity within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUREMENTS

4.6.1.6.1 Containment Tendons. The containment tendons and the containment exterior surfaces shall be examined in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants," and the modifications presented in 10 CFR 50.55a(b)(2)(viii), "Examination of concrete containments," as modified by approved exemptions. The containment structural integrity shall be demonstrated during the inspection periods specified in IWL-2410 and IWL-2420. The tendons' structural integrity shall be demonstrated by:

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.7.3 Deleted

4.7.7.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

ACTION (Continued)

startup transformer and associated circuits within one hour and at least once per 8 hours thereafter; and if the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, unless the diesel generators are already operating; restore one of the inoperable sources to OPERABLE status in accordance with Action Statements a and b, as appropriate. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the other A.C. power source (startup transformer or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

   1. All required systems, subsystems, trains, components, and devices (except safety injection pumps) that depend on the remaining required OPERABLE diesel generators as a source of emergency power are also OPERABLE.

      If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours , and in COLD SHUTDOWN within the following 30 hours.

   2. At least two Safety Injection pumps are OPERABLE and capable of being powered from their associated OPERABLE diesel generators.

      If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With two of the above required startup transformers or their associated circuits inoperable; restore at least one of the inoperable startup transformers to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours* and in COLD

*If the opposite unit is shutdown first, this time can be extended to 42 hours.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10" the spent fuel storage pool.*

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 DELETED

5.0 DESIGN FEATURES

5.1 SITE

5.1.1 The site is approximately 25 miles south of Miami, 8 miles east of Florida City, and 9 miles southeast of Homestead, Florida

5.2 DELETED

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4, ZIRLO®, or Optimized ZIRLO™ except that replacement of fuel rods by filler rods consisting of stainless steel, or by vacant rod positions, may be made in fuel assemblies if justified by cycle-specific reload analysis using NRC-approved methodology. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be clad with stainless steel tubing.

5.4 DELETED

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to, and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety, and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The plant manager shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the Health Physics Supervisor shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

PLANT STAFF

6.2.2 The plant organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. DELETED

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

f. DELETED

h. The Assistant Operations Manager - Line shall hold a Senior Reactor Operator License.

i. The Operations Manager shall either:

1. hold or have held a Senior Reactor Operator License on the Turkey Point Plant; or,

2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or

3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)

* The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
SM	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

SM - Shift Manager with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position may be filled by the SM or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

6.9.1.1 Deleted

6.9.1.2 Deleted

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall contain the following:

a. The methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and

b. The radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Annual Radioactive Effluent Release Reports required by Specification 6.9.1.3 and Specification 6.9.1.4.

6.14.2 Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR 50, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 279 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 274 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 April 9, 2017 (Reference 1), as supplemented by letter dated October 4, 2017 (Reference 2), Florida Power & Light Company (the licensee or FPL) 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 to remove certain reporting requirements, to remove the completion time for restoring spent fuel pool water level, to address inoperability of one of the two parallel flow paths in the residual heat removal or safety injection headers for the Emergency Core Cooling Systems, and to make other administrative changes, including updating and correcting a misspelling in TS 3.3.3.2, "Movable Incore Detectors."

By electronic mail (e-mail) dated August 16, 2017 (Reference 3), the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff sent the licensee a request for additional information (RAI). The licensee responded to the RAI by letter dated October 4, 2017. The licensee's 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 determination that was published in the *Federal Register* (FR) on June 19, 2017 (82 FR 27889).

2.0 REGULATORY EVALUATION

2.1. Background

2.1.1 TS Reporting Requirements

TS reporting requirements for licensees are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," as well as other parts of 10 CFR Chapter I, "Nuclear Regulatory Commission." Prior to August 2009, Regulatory Guide (RG) 1.16, "Reporting of Operating Information" (Reference 4), provided guidance to licensees on an acceptable basis for meeting the reporting requirements of the facility operating license. In particular, this RG provided a description of each of the periodic reports, including annual reports and the Startup Report that licensees are required to submit to demonstrate compliance with the TS reporting requirements. In August 2009, the NRC withdrew RG 1.16 because guidance on the content and frequency of required reports is contained in Chapter 5, "Administrative Controls," of the standard technical specifications.

2.1.2 Emergency Core Cooling System (ECCS)

In the event of a loss-of-coolant accident, emergency core cooling is provided by the Safety Injection (SI) System, which constitutes the ECCS. The principal components of the SI System are the accumulators (one for each loop), the four high-head safety injection pumps, and the two residual heat removal (RHR) pumps.

Upon initiation, the SI System isolation valves open and the high-head safety injection and the RHR pumps start. During the injection phase, both the high-head safety injection and the RHR pumps take suction from the refueling water storage tank and deliver borated-water via separate headers to all three cold legs. Each header splits into two parallel flow paths prior to connecting to the cold legs. The parallel flow paths are controlled by motor-operated valves, 744A/B for the RHR header and 843A/B for the SI header.

TS 3/4.5.2 provides the limiting condition for operation (LCO), ACTIONs, and surveillance requirements (SRs) for the ECCS when average coolant temperature (T_{avg}) is greater than or equal to 350 degrees Fahrenheit (°F).

2.2 <u>Licensee's Proposed Changes</u>

The licensee requested to delete the special reports or four-hour notifications required by the following TS:

- TS 3.1.1.3, Moderator Temperature Coefficient, ACTION a.3
- TS 3.2.4, Quadrant Power Tilt Ratio, SR 4.2.4.3
- TS 3.4.9.3, Overpressure Mitigating Systems, ACTION e
- TS 3.5.2, ECCS Subsystems – T_{avg} Greater than or Equal to 350 °F, ACTION b
- TS 3.7.7, Sealed Source Contamination, SR 4.7.7.3
- TS 3.8.1.1, A.C. Sources – Operating, ACTIONs c and e
- TS 5.3.1, Fuel Assemblies

Further, the licensee proposed to delete the engineering evaluations and special reports required by TS 3.6.1.6, Containment Structural Integrity, ACTIONs a and b, and to delete TS 6.9.1.1 and TS 6.9.1.2, which required the submittal of a Startup Report and Annual Report, respectively.

In addition to the change to TS reporting requirements discussed above, the licensee proposed to (deletions shown in stricken text and additions underlined):

TS 3.3.3.2, Movable Incore Detectors

- Correct misspelling of QUADRANT in LCO 3.3.3.2.a:

At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the ~~QUANDRANT~~ QUADRANT POWER TILT RATIO*, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

TS 3.5.2

- Revise ACTION a to include parallel injection flow paths:

~~a. With one RHR heat exchanger or suction flow path from the containment sump inoperable, restore the inoperable RHR heat exchanger or suction flow path from the containment sump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.~~

With one of the following components inoperable:

1. RHR heat exchanger,
2. RHR suction flow path from the containment sump,
3. RHR parallel injection flow path, or
4. SI parallel injection flow path

Restore the inoperable component to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TS 3.9.11, Water Level – Storage Pool

- Revise ACTION a to eliminate the four-hour completion time

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas ~~and restore the water level to within its limit within 4 hours~~.

- Eliminate the footnote that permits suspending the requirements of TS 3.9.11

~~*The requirements of this specification may be suspended for more than 4 hours to perform maintenance provided a 10 CFR 50.59 evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.~~

TS 6.1, Responsibility; TS 6.2, Organization

- Revise the title Plant General Manager:

TS 6.1.1 The ~~Plant General Manager~~ plant manager shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

TS 6.2.1.c The ~~Plant General Manager~~ plant manager shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

- Revise TS 6.1.2 regarding control room command function:

TS 6.1.2 ~~The Nuclear Plant Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.~~
The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

- Revise Table 6.2-1 regarding minimum shift crew composition:

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
SM ~~NPS~~	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

SM ~~NPS~~ – Shift Manager ~~Nuclear Plant Supervisor~~ with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

~~During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.~~

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* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position may be filled by the <u>SM</u> ~~Nuclear Plant Supervisor~~ or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

- Revise TS 6.2.2 and Table 6.2-1 to change titles:

  TS 6.2.2.h. The <u>Assistant</u> Operations <u>Manager - Line</u> ~~Supervisor~~ shall hold a Senior Reactor Operator License.

<u>TS 6.14, Offsite Dose Calculation Manual (ODCM)</u>

- Revise the title Plant General Manager in TS 6.14.2.b:

  Shall become effective after approval of the ~~Plant General Manager~~ <u>plant manager</u>; and

Finally, the licensee requested to make administrative updates to the TS index and affected TSs based on the changes discussed above.

2.3 <u>Regulatory Review</u>

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.

Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 50.36(a)(1) 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."

Section 50.55a of 10 CFR, "Codes and standards," establishes the applicable codes and standards for systems, structures, and components at nuclear power plants.
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 50.55a(b)(2)(viii) of 10 CFR requires, in part, that containment tendons and containment exterior surfaces be examined in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants."

Section 50.67 of 10 CFR, "Accident source term," sets limits for the radiological consequences of postulated design-basis accidents using an alternative source term. Paragraph 50.67(b)(2) of 10 CFR 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.

Section 50.72 of 10 CFR, "Immediate notification requirements for operating nuclear power reactors," establishes the requirements for licensee notification of the NRC via the Emergency Notification System.

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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.

Section 50.73 of 10 CFR, "Licensee event report system," establishes the requirements for licensees to provide to the NRC 60-day written licensee event reports.

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 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 No. 3) and 240 (Unit No. 4), dated June 23, 2011 (Reference 5).

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 No. 3) and 240 (Unit No. 4) used an AST methodology for analyzing the radiological consequences of eight design-basis accidents using 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.

Guidance

NUREG-0800, "Standard Review Plan [SRP] 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, July 2000 (Reference 6), 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-1022, Revision 3, "Event Report Guidelines: 10 CFR 50.72 and 10 CFR 50.73," dated January 2013 (Reference 7), contains guidelines that the NRC staff considers acceptable for use in meeting the requirements of 10 CFR 50.72 and 10 CFR 50.73.

NUREG-1431, Revision 4, "Standard Technical Specifications [STS] – Westinghouse Plants" (Reference 8) 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 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (Reference 9) 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.

Licensing and Design Basis Information

The NRC staff also considered relevant information in the Turkey Point 3 and 4 UFSAR (Reference 10).

3.0 TECHNICAL EVALUATION

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

3.1 Evaluation of Proposed Changes to TS Reporting Requirements

TS 3.1.1.3 Moderator Temperature Coefficient

Turkey Point 3 and 4 TS 3.1.1.3 ACTION a.3 currently states:

> A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured [moderator temperature coefficient], the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive [moderator temperature coefficient] to within its limit for the all rods withdrawn condition.

The licensee proposes to delete the above TS reporting requirement and to make a corresponding editorial change to ACTION a.2 to reflect the deletion of ACTION a.3. The report provides information to the NRC, but does not seek any approval nor does it ensure the safe operation of the facility during, or after, the 10 days provided to submit the report. Should the moderator temperature coefficient be more positive than the beginning-of-cycle-life limit specified in the Core Operating Limits Report, the remaining ACTIONs (a.1 and a.2) require that: (1) control rod withdrawal limits be established and maintained sufficient enough to restore the moderator temperature coefficient to less positive or equal to the beginning-of-cycle-life limit specified in the Core Operating Limit Report, and (2) the control rods be maintained within the withdrawal limits established, until a subsequent calculation verifies that the moderator temperature coefficient has been restored to within its limit for the all-rods-withdrawn condition.

The NRC staff finds that this change is consistent with STS 3.1.3, "Moderator Temperature Coefficient (MTC)," in NUREG-1431. Further, the NRC staff finds that TS 3.1.1.3 ACTIONs a.1 and a.2 are sufficient to provide reasonable assurance that the moderator temperature coefficient will be restored to within limits for the all-rods-withdrawn condition. As such, the licensee will continue to meet the requirements of 10 CFR 50.36(c)(2). Therefore, the NRC staff concludes that it is acceptable to delete TS 3.1.1.3 ACTION a.3. The corresponding change to

ACTION a.2 to reflect the deletion of ACTION a.3 is editorial in nature and is, therefore, acceptable.

TS 3.2.4 Quadrant Power Tilt Ratio

Turkey Point 3 and 4 SR 4.2.4.3 currently states:

> If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report in accordance with 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

The licensee proposes to delete the above TS reporting requirement. The regulations at 10 CFR 50.36(c)(3) state that SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. TS 3.2.4 ACTIONs a.3 and a.4 currently state:

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

These ACTIONs in TS 3.2.4 ensure that the facility is operating within safety limits by requiring a power reduction to a level outside the applicability of TS 3.2.4, if the Quadrant Power Tilt Ratio is not restored to within limits in 24 hours. Submitting a special report is not necessary to meet the requirements of 10 CFR 50.36(c)(3). Also, the proposed change is consistent with STS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," in NUREG 1431, which does not require a special report. The licensee stated that the condition in which the Quadrant Power Tilt Ratio exceeds its limit for greater than 24 hours would be entered into the corrective action program, which is subject to NRC review and audit.

The NRC staff finds that SRs 4.2.4.1 and 4.2.4.2 are sufficient to provide reasonable assurance that TS 3.2.4 will be met. The NRC staff further finds that the reporting requirements in SR 4.2.4.3 are not needed to meet the requirements in 10 CFR 50.36(c)(3). Therefore, the NRC staff concludes that it is acceptable to delete SR 4.2.4.3.

TS 3.4.9.3, Overpressure Mitigating Systems

Turkey Point 3 and 4 TS 3.4.9.3, ACTION e, currently states:

> In the event either the PORVs [power-operated relief valves] or a 2.20 square inch vent is used to mitigate an RCS [reactor coolant system] pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to

> Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

The licensee proposes to delete the above TS reporting requirement. The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI [Three Mile Island]-Related Requirements for New Operating Licenses." The guidance of NUREG-0694 states that licensees shall "[a]ssure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." Subsequently, the NRC issued Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" (Reference 11) requesting the submittal of less information in the monthly operating report. As a result, the NRC-approved TSTF-258, "Changes to Section 5.0, Administrative Controls," deleted the requirement to provide documentation of all challenges to the PORVs and safety valves. The NRC's regulations at 10 CFR 50.73 require a licensee to submit a Licensee Event Report (LER) for any event of the type described in the regulation within 60 days after the discovery of the event. The information needed by the NRC staff to evaluate a challenge to a PORV is adequately addressed by the reporting requirements in this regulation. Further, the licensee stated that a low temperature over-pressure transient that violates the TS pressure-temperature limits would be immediately reportable under 10 CFR 50.72.

The NRC staff finds that the reporting requirements in 10 CFR 50.72 and 10 CFR 50.73 adequately address the reporting of low temperature over-pressure transients. Therefore, the NRC staff concludes that deletion of the reporting requirement, TS 3.4.9.3, ACTION e, is acceptable.

As a result of deleting TS 3.4.9.3, ACTION e, the licensee proposes to redesignate ACTION f to ACTION e. This change is administrative in nature and is, therefore, acceptable.

<u>TS 3.5.2, ACTION b and TS 3.5.3, ACTION c - ECCS</u>

Turkey Point 3 and 4 TS 3.5.2, ACTION b and TS 3.5.3, ACTION c, currently state:

> In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

The licensee proposes to delete the above TS reporting requirements based on the requirements in 10 CFR 50.73(a)(2)(iv) and 10 CFR 50.72(b)(2)(iv). Specifically, 10 CFR 50.73(a)(2)(iv) requires a licensee, with some exceptions, to submit an LER in the event of an ECCS actuation. Additionally, in accordance with 10 CFR 50.73(a), the LER must be submitted within 60 days after the discovery of the event. The regulations at 10 CFR 50.72(b)(2)(iv) require licensees to notify the NRC within 4 hours for an ECCS actuation that results in, or should have resulted in, ECCS discharge into the RCS in response to a valid signal. This requirement excludes actuations that result from, and are part of, a pre-planned sequence during testing or reactor operation.

Currently, TS 3.5.2, ACTION b, and TS 3.5.3, ACTION c, require the licensee to provide the "total accumulated actuation cycles to date since January 1, 1990" as part of any Special Report, which is not required to be included as part of an LER submitted in accordance with the regulations at 10 CFR 50.73. The purpose of these TS requirements is to provide assurance of the integrity of the components in the reactor coolant pressure boundary for the design life of the plant due to thermal cycles (i.e., injection of water into the reactor vessel by the ECCS results in a thermal cycle). As discussed in Turkey Point 3 and 4 UFSAR Section 4.1.5, "Cyclic Loads," Table 4.1-10, "Component Cyclic or Transient Limits," provides the component cyclic or transient limits for the RCS and secondary coolant system previously relocated from TS 5.6. Turkey Point 3 and 4 procedure 0-ADM-553, "Maintaining Records for Design Cycles," establishes the requirement for recording the actual cycles imposed on various plant systems to ensure that the limits specified in UFSAR Table 4.1-10 are not exceeded. The licensee stated that corrective action is taken if any plant cycle comes within 80 percent of the design cycle limit.

The NRC staff finds that the reporting requirements in 10 CFR 50.73, the limits in UFSAR Section 4.1.5, Table 4.1-10, and the monitoring provided by the Turkey Point 3 and 4 surveillance procedures noted above, provide reasonable assurance that ECCS actuations will be appropriately tracked consistent with assuring that the integrity of the components in the reactor coolant pressure boundary will be maintained within design limits. As such, the licensee will continue to meet the requirements of 10 CFR 50.36(c)(2). Therefore, the NRC staff concludes that deletion of TS 3.5.2, ACTION b and TS 3.5.3, ACTION c, is acceptable.

TS 3.6.1.6, Containment Structural Integrity

Turkey Point 3 and 4 TS 3.6.1.6, ACTIONs a, b, and c, currently state:

a. With more than one tendon with an observed lift-off force between 90% and 95% of the predicted force, or with one tendon below 90% of the predicted force, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the average of all measured tendon forces for each type of tendon (dome, vertical, and hoop), including those measured in ACTION a., less than the predicted force, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any abnormal degradation of the structural integrity other than ACTION a. and ACTION b., at a level below the acceptance criteria of Specifications 4.6.1.6.1, 4.6.1.6.2 and 4.6.1.6.3, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The licensee proposes to delete the requirement to perform an engineering evaluation and submit a Special Report to the NRC from the ACTIONs above.

The containment tendons and the containment exterior surfaces are examined in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition with the 2003 Addenda, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants," and the modifications presented in 10 CFR 50.55a(b)(2)(viii), "Section XI condition: Concrete containment examinations." The containment structural integrity shall be demonstrated during the inspection periods specified in IWL-2410 and IWL-2420. The regulation covering the inservice inspection of the containment tendons is 10 CFR 50.55a(g)(4). This regulation requires, in part, that throughout the service of a pressurized water-cooled nuclear power facility, components that are classified as ASME Code Class CC, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Code and Addenda that are incorporated by reference in paragraph (a)(1)(ii) of 10 CFR 50.55a, subject to the limitations and modifications listed in paragraph (b)(2).

All reportable events are addressed under Subsections IWA and IWL of the ASME Code, as required by 10 CFR 50.55a. Because the reporting requirements are in accordance with 10 CFR 50.55a(b)(2)(viii), the requirements in TS 3.6.1.6 ACTIONs a, b, and c, to perform an engineering evaluation and submit a Special Report to the NRC are no longer necessary. As such, the NRC staff finds that these requirements in TS 3.6.1.6, ACTIONs a, b, and c, are not needed to meet the requirements in 10 CFR 50.36(c)(2). On this basis, the NRC staff concludes that deletion of these requirements from TS 3.6.1.6 is acceptable.

TS 3.7.7, Sealed Source Contamination

Turkey Point 3 and 4 SR 4.7.7.3 currently states:

> Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

The licensee proposes to delete the reporting requirements contained in SR 4.7.7.3. The requirement for SRs is provided in the regulations at 10 CFR 50.36(c)(3), which states that SRs "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The LCO associated with SR 4.7.7.3, LCO 3.7.7, requires that:

> Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

The ACTION statement for TS 3.7.7 states, in part:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

    1. Decontaminate and repair the sealed source, or

2. Dispose of the sealed source in accordance with Commission Regulations.

SRs 4.7.7.1 and 4.7.7.2 provide the test requirements and test frequencies, respectively, to ensure that the LCO requirements are met. With respect to test requirements, SR 4.7.7.1 states that, "[e]ach sealed source shall be tested for leakage and/or contamination" and that the method of testing shall have a detection sensitivity of at least 0.005 microCuries per test sample. With respect to test frequencies, SR 4.7.7.2, provides specific test frequencies for sources in use, stored sources not in use, and startup sources and fission detectors. Also, should the licensee identify a source exceeding the allowable limits, it would enter the source into the corrective action program, which is subject to inspection by the NRC staff.

The NRC staff finds that SRs 4.7.7.1 and 4.7.7.2 are sufficient to provide reasonable assurance that LCO 3.7.7 will continue to be met after the proposed change. The NRC staff further finds that because the reporting requirements in SR 4.7.7.3 do not relate to tests, calibrations, or inspections, they are not appropriate to meet the requirements in 10 CFR 50.36(c)(3). Further, failed leak tests are not required to be reported by 10 CFR 50.72. Therefore, the NRC staff concludes that it is acceptable to delete SR 4.7.7.3.

<u>TS 3.8.1.1, A.C. Sources – Operating</u>

Turkey Point 3 and 4 TS 3.8.1.1, ACTION c, currently states, in part:

> Notify the NRC within 4 hours of declaring both a start-up transformer and diesel generator inoperable.

Turkey Point 3 and 4 TS 3.8.1.1, ACTION e, currently states, in part:

> With two of the above required startup transformers or their associated circuits inoperable notify the NRC within 4 hours.

The licensee proposes to delete the above TS reporting requirements. TS 3.8.1.1 requires operable AC power sources consisting of three emergency diesel generators and two startup transformers. However, NUREG-1431 and NUREG-1022 do not include such reporting requirements for inoperable AC power sources. The NRC's requirements for submitting four-hour reports are contained in the regulations at 10 CFR 50.72(b)(2). The inoperability of a portion of a plant's required AC sources does not meet the reporting criteria. For a more severe condition, such as a condition that results in a TS-required plant shutdown or the declaration of an emergency class, the degradation of AC power sources would be immediately reportable in accordance with 10 CFR 50.72. Similarly, the inoperability of AC power sources that results in the inability to fulfill a safety function would be reportable in accordance with 10 CFR 50.72 and 10 CFR 50.73.

The NRC staff finds the proposed changes to be consistent with the immediate reporting criteria specified in 10 CFR 50.72. Additionally, the NRC staff finds the proposed changes to be consistent with both NUREG-1431 and NUREG-1022. The NRC staff further finds that the reporting requirements in TS 3.8.1.1, ACTIONs c and e, are not needed to meet the requirements in 10 CFR 50.36(c)(2). Therefore, the NRC staff concludes that the proposed changes, deletion of the four-hour reports from TS 3.8.1.1, ACTIONs c and e, are acceptable.

TS 5.3.1, Fuel Assemblies

Turkey Point 3 and 4 TS 5.3.1 currently states, in part:

> Should more than 30 individual rods in the core, or 10 fuel rods in any fuel assembly, be replaced per refueling, a Special Report discussing the rod replacements shall be submitted to the Commission within 30 days after cycle startup.

The licensee proposes to delete the above TS reporting requirement. Each cycle-specific core reload is implemented using NRC-approved methodologies and the licensee's design change process. Section 3.2.1, "Nuclear Design and Evaluation," in the UFSAR states that cycle-specific values are calculated and their impact on plant and safety analyses is evaluated prior to each cycle. The results of these evaluations for the current cycles are presented in UFSAR Appendices 14A and 14B for Unit Nos. 3 and 4, respectively. The cycle-specific design parameters for the current reloads are documented in nuclear design reports that are used for design verification and operational guidance. Information regarding cycle-specific changes made to core design during refueling, which includes documentation prepared in accordance with the regulations at 10 CFR 50.59, is available for inspection by the NRC staff. In addition, STS 4.2.1, "Fuel Assemblies," does not require licensees to prepare a similar report.

The NRC staff finds that the proposed change is consistent with NUREG-1431. Further, cycle-specific values and impact on plant safety analyses for reloads are available in UFSAR Appendices 14A and 14B and information regarding core design changes is available for inspection by the NRC staff. Therefore, the NRC staff concludes that the licensee will continue to meet the requirements of 10 CFR 50.36(c)(2) and that the proposed change, deletion of the requirement to submit a 30-day report, is acceptable.

TS 6.9.1.1, Startup Report

Turkey Point 3 and 4 TS 6.9.1.1 currently states:

> A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.
>
> The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.
>
> Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever

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

The licensee proposes to delete the above TS reporting requirements pertaining to the Startup Report. The NRC's regulation at 10 CFR 50.36(c)(5) states, that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." In general, Turkey Point 3 and 4 TS Section 6.0, "Administrative Controls," contains those requirements not covered by other TS sections, but which are necessary to assure the safe operation of the facility.

In October 1971, the U.S. Atomic Energy Commission, the predecessor to the NRC, published RG 1.16 (formerly Safety Guide 16), "Reporting of Operating Information." The purpose of the RG was to provide an acceptable basis for implementing the reporting requirements of 10 CFR and the TSs. The Startup Report information in the RG is nearly identical to that in Turkey Point 3 and 4 TS 6.9.1.1.

On August 11, 2009 (74 FR 40244), the NRC withdrew RG 1.16, because it was no longer needed. Specifically, the notice stated that reporting requirements are contained in 10 CFR Part 50, as well as other parts of 10 CFR Chapter I. In addition, the *Federal Register* notice stated that guidance on the content and frequency of required reports is provided in Chapter 5, "Administrative Controls," of the STS contained in the applicable NUREG (i.e., NUREG-1431 for Turkey Point 3 and 4).

Turkey Point 3 and 4 TS 6.9.1.1 provides the time frames for submittal of the reports (i.e., 90 days following completion of the startup test program, 90 days following resumption or commencement of commercial power operation, or 9 months following initial criticality). Given these time frames, report completion and submittal are not necessary to assure safe operation of the facility for the time frame between completion of the testing and submittal of the report. In addition, there is no requirement for NRC approval of the information provided in the report after it is submitted. In essence, these reports merely provide the NRC a mechanism to review the appropriateness of the licensee activities after-the-fact.

The NRC staff has determined that the proposed elimination of the Startup Report requirements would not eliminate the need to perform the necessary testing. Appropriate startup testing and documentation of startup testing will continue to be performed in accordance with the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criterion XI, "Test Control." Specifically, Criterion XI requires that testing be performed to demonstrate that structures, systems, and components will perform satisfactorily in service and that test results be documented. The licensee stated that information provided in the plant startup report is readily available to the NRC for inspection. Based on the testing, documentation and retention requirements stated above, the NRC staff may review the test results onsite, as needed.

On the basis that the Startup Report requirements in TS 6.9.1.1 are not necessary to assure operation of the facility in a safe manner, the NRC staff finds that deletion of these TS requirements is consistent with 10 CFR 50.36(c)(5). Therefore, the NRC staff concludes that deletion of TS 6.9.1.1 is acceptable. As a result of this change, the licensee proposes to update the TS Index page to reflect the deletion of TS 6.9.1.1. This change is editorial in nature and is, therefore, acceptable.

TS 6.9.1.2, Annual Report

Turkey Point 3 and 4 TS 6.9.1.2 and associated footnote currently states:

> ANNUAL REPORTS*
>
> 6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.
>
> Reports required on an annual basis shall include:
>
> The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 0.25 microcuries per gram DOSE EQUIVALENT I-131.
>
> * A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

The licensee proposes to delete the above TS reporting requirements pertaining to the Annual Report, including its associated footnote. The above TS reporting requirement is based on the model TSs provided in Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes," dated September 27, 1985 (Reference 12).

The licensee's application stated that specific activity analysis pertaining to primary coolant limits is reported to the NRC by means of the Performance Indicator Program, under the Reactor Oversight Process. As part of this program, the licensee provides to the NRC, on a quarterly basis, monthly reactor coolant specific activity data. This is done in accordance with Regulatory Issue Summary 2000-08, Revision 1, "Voluntary Submission of Performance Indicator Data" (Reference 13) and follows the guidelines provided in Nuclear Energy Institute 99-02, Revision 6, "Regulatory Assessment Performance Indicator Guideline," dated October 2009 (Reference 14). The reactor coolant specific activity concentration is provided more frequently than that required by the current TS, regardless of whether or not the TS limit is exceeded.

As discussed in Section 2.3 of Nuclear Energy Institute 99-02, the purpose of the performance indicator related to RCS specific activity is to monitor the integrity of the fuel cladding (one of the barriers to prevent release of fission products). In accordance with the regulations at 10 CFR 50.72, licensees are required to provide notification to the NRC within 8 hours regarding "[t]he condition of the nuclear power plant, including its principal safety barriers, being seriously degraded...." The same condition needs to be reported to the NRC within 60 days via an LER, in accordance with 10 CFR 50.73.

In addition to the above reporting requirements, Turkey Point 3 and 4 TS LCO 3.4.8, "Specific Activity," puts limitations on the specific activity of the reactor coolant. These limitations ensure that in the event of a release of any radioactive material to the environment during a design-basis accident, radiation doses are maintained within the limits of 10 CFR Part 100. LCO 3.4.8 requires that the specific activity of the primary coolant be limited to less than or equal to 0.25 microcuries per gram dose equivalent iodine 131. In the event that the specific activity of the primary coolant exceeds the limits of LCO 3.4.8, the TS ACTIONs require, in part, that the licensee perform sampling and analysis until the specific activity is restored to within its limit.

Based on the reporting requirements in 10 CFR 50.72 and 50.73, the specific activity reporting provided under the Performance Indicator Program, and the requirements in TS LCO 3.4.8, the NRC staff finds that the reporting requirements in TS 6.9.1.2 and its associated footnote are not necessary to assure operation of the facility in a safe manner. Therefore, the NRC staff finds that deletion of this TS is consistent with 10 CFR 50.36(c)(5). On this basis, the NRC staff concludes that deletion of TS 6.9.1.2 and its associated footnote are acceptable. As a result of this change, the licensee proposes to update the TS Index page to reflect the deletion of TS 6.9.1.2. This change is editorial in nature and is, therefore, acceptable.

3.2 Evaluation of Proposed Change to TS LCO 3.3.3.2.a

The current Turkey Point 3 and 4 TS LCO 3.3.3.2.a states:

> At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the QUANDRANT POWER TILT RATIO*, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

The licensee proposes to replace the word, "QUANDRANT" with "QUADRANT."

The proposed change corrects a misspelling in TS LCO 3.3.3.2.a. The proposed change is editorial in nature and will not affect the licensee's ability to continue to meet the requirements of 10 CFR 50.36(c)(2). Therefore, the proposed change is acceptable.

3.3 Evaluation of Proposed Changes to TS 3.5.2, ACTION a

The current TS 3.5.2, ACTION a, became effective on November 15, 2015, through issuance of Amendment Nos. 267 and 262 for Turkey Point 3 and 4, respectively (Reference 15). Prior to these Amendments, ACTION a (Reference 16) required that:

> With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s) or an inoperable RHR pump, restore the inoperable component or flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

The above ACTION a addressed inoperable ECCS flow paths and allowed 72 hours to restore an inoperable flow path to operable condition before a plant shutdown would be required. Following the amendments, the "ECCS flow paths" was replaced with "suction flow path from the containment sump." The licensee identified that the current ACTION a only addresses the section of the RHR discharge flow path associated with a particular RHR pump and it does not account for the parallel flow paths, which are not associated with a particular SI or RHR pump. To correct the TS omission, the licensee proposes to revise ACTION a by adding the RHR parallel injection flow path, and the SI parallel injection flow path, as follows:

With one of the following components inoperable:

1. RHR heat exchanger,
2. RHR suction flow path from the containment sump,
3. RHR parallel injection flow path, or
4. SI parallel injection flow path

Restore the inoperable component to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Isolation Valves 744A/B and 843A/B

Figures in Chapter 6.2 of the Turkey Point 3 and 4 UFSAR indicate that the RHR and SI cold-leg headers each contain two parallel flow paths with a motor-operated isolation valve in each flow path. Those four isolation valves (744A/B and 843A/B) would open on an SI actuation signal. While the proposed TS would add the RHR parallel injection flow paths and SI parallel injection flow paths to ACTION a of TS 3.5.2, limiting the outage time of the associated flow paths to 72 hours, the valves 744A/B and 843A/B were not added to TS SR 4.5.2.a. SR 4.5.2.a specifies the requirements for applicable valves in the ECCS flow paths for determining if the requirements in TS 3.5.2, ACTION a, are met for a corresponding flow path. In accordance with the requirements in 10 CFR 50.36(c)(3), which requires, in part, establishing SRs to assure that the LCOs will be met, the NRC staff issued a request for additional information (RAI) requesting the licensee to provide a justification for the acceptability of not including valves 744A/B and 843 A/B in TS SR 4.5.2.a.

In response to the RAI, the licensee indicated that the high-head cold-leg injection valves (843A/B) and the low-head cold-leg injection valves (744A/B) are normally closed and energized, motor-operated valves that automatically open upon receipt of an SI signal. Including these valves in TS SR 4.5.2.a would be inappropriate because this SR requires that certain valves be in specified positions with power to the valve operators removed. Removing power from 744A/B and 843A/B cold-leg injection valves would make them incapable of performing their required function to automatically open on an actuation signal. There are other SRs in current TS 3.5.2 to verify the operability of the 744A/B and 843A/B valves: (1) SR 4.5.2.b.2, which requires the licensee to verify that each valve (manual, power-operated, or automatic) in the ECCS flow path that is not locked, sealed, or otherwise secured in position, is in its correct position; and (2) SR 4.5.2.f.1, which requires the licensee to verify that these valves actuate to their correct position on an SI actuation test signal. These SRs are equivalent to SR 3.5.2.2 and SR 3.5.2.5 in NUREG-1431. Therefore, the NRC staff agrees with the licensee that current TS 3.5.2 includes appropriate SRs for the ECCS cold-leg injection valves consistent with 10 CFR 50.36(c)(3) to demonstrate that the LCOs in TS 3.5.2 will be met.

RCS Hot-Leg Recirculation Flow Paths

Chapter 6.2.2 of the UFSAR indicates that the RCS hot-leg injection piping configuration contains two parallel injection flow paths with a motor-operated isolation valve in each flow path. The RCS hot-leg injection is required to prevent boric acid-precipitation on the fuel cladding from reducing core cooling following a loss-of-coolant accident (LOCA). Those two isolation valves (866A/B) would open during the RCS hot-leg injection mode, and they are included in TS SR 4.5.2.a, which specifies the requirements for applicable valves to assure ECCS flow paths are operable. The hot-leg parallel injection follow paths are not included in the TS 3.5.2 LCO and associated ACTIONs.

According to Criterion 3 of 10 CFR 50.36(c)(2)(ii) a TS LCO is required for a "structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The NRC staff issued an RAI requesting the licensee to provide a justification for the acceptability of not including the RCS hot-leg injection flow paths in the TS 3.5.2 LCO and associated ACTIONs.

In its response to the RAI, the licensee stated that:

> The hot leg recirculation flow path via the SI pumps is provided for continuation of the recirculation phase. A RHR train supplies flow to the SI pumps suctions. The SI pumps discharge to the hot leg injection header. Hot leg recirculation is required to prevent boric acid plate-out on the fuel cladding from reducing core cooling following a cold leg break.
>
> To avoid the potential steam binding due to injection into the hot legs early in any LOCA transient when steam generators are still relatively hot, the 866 valves which control the flow paths to the hot legs are maintained closed by keeping the motor circuit breakers locked open at the motor control centers. This administrative control ensures that automatic or inadvertent manual actions do not result in hot leg injection. As such, manual action is required to energize the 866 valves and initiate hot leg injection.
>
> The requirement in TS LCO 3.5.2.a for the SI pump discharge flow paths to be aligned to the RCS cold legs was incorporated on August 28, 1990 by License Amendments 137 and 132 for Units 3 and 4, respectively. This alignment is associated with the initial automatic actuation of SI and occurs without operator action on a SI signal.
>
> License Amendments 137 and 132 [Reference 17] replaced the former Turkey Point custom TS with TS based on Draft Revision 5 of NUREG-0452 (the original Westinghouse Standard Technical Specifications) within certain limitations. Those limitations were that the revised TS would not require hardware changes, would reflect the current plant design and analytical basis, and would consider operating hardship or reasonable additions. A TS LCO specific to the hot leg flow paths was not incorporated.
>
> 10 CFR 50.36(c)(2)(iii) states: "A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section." 10 CFR 50.36(c)(2)(iii) applies to TS LCO 3.5.2.a because the specified alignment to the cold legs was established prior to August 18, 1995.
>
> FPL adequately maintains the 866 valves functional as follows:
>
> - The 866 valves are included in the Inservice Testing Program and are functionally tested when the units are in cold shutdown and the RCS is depressurized if the valves have not been tested within the previous 92 days.

- The 866 valves are verified to be closed with power removed in accordance with TS SR 4.5.2.a. They are also verified to be closed with power removed during the performance of TS SR 4.5.2.b.2.
- The 866 valves are included in the scope of the Maintenance Rule as required by 10 CFR 50.65(b)(1).

Based on the above, 10 CFR 50.36(c)(2)(iii) does not require FPL to propose a change to TS LCO 3.5.2.a. In addition, current FPL testing and maintenance processes ensure that the 866 valves are maintained to a high degree of availability.

The NRC staff agrees with the licensee that 10 CFR 50.36(c)(2)(iii) is applicable to TS LCO 3.5.2.a, because the specified alignment to the cold legs was established prior to August 18, 1995. Therefore, TS 3.5.2, ACTION a does not need to be modified to include the hot-leg injection flow paths.

Conclusion for Changes to TS 3.5.2, ACTION a

The proposed TS 3.5.2 ACTION a is intended to address inoperability of the parallel flow paths in the RHR and SI injection headers and to provide a 72-hour completion time for restoration of an inoperable flow path. The NRC staff finds that the changes establish a TS ACTION for an inoperable RHR or SI parallel flow path that existed prior to Amendments 267 and 262. As such, the NRC staff finds that the proposed change restores the more restrictive operability requirements and is acceptable.

For each unit, the SI and RHR pumps in conjunction with the accumulators are assumed in the LOCA analysis to supply sufficient core cooling to meet the performance acceptance criteria of the ECCS in 10 CFR 50.46 for a postulated LOCA. The NRC staff finds that with the acceptable TS 3.5.2 ACTION a, the licensee meets Criterion 3 of Paragraph 50.36(c)(2)(ii) that requires, in part, a TS LCO for a structure, system, or component that is credited for the design basis analyses, including a LOCA analysis.

3.4 Evaluation of Proposed Changes to TS 3.9.11

The licensee proposes to modify Turkey Point 3 and 4 TS 3.9.11, ACTION a, and to delete a footnote for TS LCO 3.9.11.

Change to ACTION a

The licensee stated that the required minimum pool level of TS LCO 3.9.11 ensures that sufficient shielding would be available during fuel movement and that adequate retention of iodine would occur in the event of a fuel handling accident. The minimum pool level is consistent with the safety analysis assumption that a minimum water depth of 23 feet would be maintained over any damaged fuel assembly.

The required ACTION of TS 3.9.11 contains the following two actions:

- suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and

- restore the water level to within its limit within 4 hours.

The proposed change would delete the second required action to restore the water level to within its limits within 4 hours. The first requirement to suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas effectively precludes fuel damage from occurring because items that could have sufficient energy to damage fuel have been placed in a condition where further movement is not credible. The action to restore storage pool level to within its limit is not necessary to satisfy the fuel handling accident analysis, because the fuel handling accident is not credible once the first action has been completed. As such, the proposed change maintains the inputs and assumptions of the radiological consequence design-basis accident analyses, and the NRC staff finds these proposed changes to be acceptable from a radiological dose perspective. Therefore, the NRC staff finds the deletion of the portion of the required action specifying restoration of the storage pool water level to within its limit within 4 hours acceptable because the action is not necessary to conform to the accident analysis assumptions, consistent with the requirements of 10 CFR 50.36(b).

Deletion of Footnote for TS LCO 3.9.11

The footnote proposed for deletion permits suspension of the requirements of TS LCO 3.9.11 for maintenance under certain conditions. This footnote states:

> *The requirements of this specification may be suspended for more than 4 hours to perform maintenance provided a 10 CFR 50.59 evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

The footnote is not necessary to permit maintenance, because the portion of the required action specifying restoration of water level within 4 hours is being deleted. As a result, maintenance activities can continue indefinitely provided that all movement of fuel assemblies and crane operations with loads in the fuel storage areas have been suspended. The proposed change does not modify any plant equipment nor does it change operation of any plant equipment. In addition, these proposed changes maintain the inputs and assumptions of the radiological consequence design-basis accident analyses. Therefore, the NRC staff finds these proposed changes to be acceptable from a radiological dose perspective. Further, because the suspension of these activities effectively precludes a fuel handling accident, the temporary reduction in storage pool level is acceptable because the water level is not necessary to satisfy fuel handling accident analysis assumed initial conditions, consistent with 10 CFR 50.36(b). With accident analysis assumptions satisfied, the report to the NRC is not necessary because operation of the facility remains safe and consistent with the fuel handling accident analysis. Therefore, deletion of the footnote is acceptable.

3.5 Evaluation of Proposed Changes to Plant Staff and Responsibilities

Changes Related to Plant General Manager Title

Turkey Point 3 and 4 TS 6.1.1, TS 6.2.1.c, and TS 6.14.2.b currently state:

> TS 6.1.1 The Plant General Manager shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

TS 6.2.1.c The Plant General Manager shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

TS 6.14.2.b Shall become effective after approval of the Plant General Manager; and

The licensee proposes to change the plant-specific title, "Plant General Manger," with the generic position title of, "plant manager."

The proposed changes are administrative in nature as they do not change the responsibility of the person designated for overall plant operations or for approval of changes to the Offsite Dose Calculation Manual. Further, the proposed changes are consistent with STS 5.1.1, STS 5.2.1.c, and STS 5.5.1 in NUREG-1431. Therefore, the proposed changes are acceptable because they are consistent with the requirements of 10 CFR 50.36(c)(5).

<u>Changes to TS 6.1.2</u>

Turkey Point 3 and 4 TS 6.1.2 currently states:

> The Nuclear Plant Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

The licensee proposes to change TS 6.1.2 to state:

> The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

The list of definitions in Table 6.2-1 currently contains the following definition:

> NPS – Nuclear Plant Supervisor with a Senior Operator license

The licensee proposes to replace the position title, "Nuclear Plant Supervisor," with that of "Shift Manager," and to update the table and footnote *** with the replacement acronym, "SM." In addition, the licensee proposes to delete the following note from Table 6.2-1:

> During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

The proposed changes are administrative in nature as they do not functionally change the responsibility of the person designated for control room command function. Further, the proposed change to TS 6.1.2 makes it consistent with STS 5.1.2 in NUREG-1431 and the note proposed for deletion in Table 6.2-1 is no longer necessary based on the proposed change to TS 6.1.2. Therefore, the proposed changes are acceptable because they are consistent with the requirements of 10 CFR 50.36(c)(5).

Change to TS 6.2.2.h

Turkey Point 3 and 4 TS 6.2.2.h currently states:

> The Operations Supervisor shall hold a Senior Reactor Operator License.

The licensee proposes to change "Operations Supervisor" to "Assistant Operations Manager – Line."

The proposed change is administrative in nature and does not involve a change in responsibility. Therefore, the proposed change is acceptable because it is consistent with the requirements of 10 CFR 50.36(c)(5).

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 February 15, 2018 (Reference 18), 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 or change 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 June 19, 2017 (82 FR 27889), that the amendments involve no significant hazards consideration, 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 considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

7.0 REFERENCES

1 Summers, Thomas, Florida Power & Light Company, letter to U.S. Nuclear Regulatory Commission, "License Amendment Request 249: Elimination of Certain Technical Specification Reporting Requirements, Revised Action for Emergency Core Cooling System, and Changes to Administrative Technical Specifications," dated April 9, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17101A637).

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3 Wentzel, Michael, U.S. Nuclear Regulatory Commission, email to Mitch Guth, Florida Power & Light Company, "Request for Additional Information - Turkey Point 3 & 4 LAR-249 (CAC Nos. MF9601 & MF9602)," dated August 16, 2017 (ADAMS Accession No. ML17228A716).

4 U.S. Atomic Energy Commission, Regulatory Guide 1.16, Revision 1 "Reporting of Operating Information," October 1973 (ADAMS Accession No. ML13226A218).

5 Paige, Jason, U.S. Nuclear Regulatory Commission, letter to Mano Nazar, Florida Power & Light Company, "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Alternative Source Term (TAC Nos. ME1624 and ME1625)," dated June 23, 2011 (ADAMS Accession No. ML110800666).

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7 U.S. Nuclear Regulatory Commission, NUREG-1022, "Event Report Guidelines 10 CFR 50.72 and 50.73," Revision 3, January 2013 (ADAMS Accession No. ML13032A220).

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17 Edison, Gordon, U.S. Nuclear Regulatory Commission, letter to Goldberg, J.H., Florida Power & Light Company, "Turkey Point Units 3 and 4 – Issuance of Amendments Re: Replacement of Current Technical Specifications with Revised Technical Specifications (TAC Nos. 63038 and 63039, 55915 and 55916, 55384 and 55385, 71864 and 71865)," dated August 28, 1990 (ADAMS Accession No. ML013440606).

18 Wentzel, Michael, U.S. Nuclear Regulatory Commission, email to Becker, Cindy, Florida Department of Health, "NRC Notification of State of Florida Regarding Turkey Point 3 and 4 License Amendments - Elimination of Certain Technical Specification Reporting Requirements," dated February 15, 2018 (no response received by February 28, 2018) (ADAMS Accession No. ML18059A819).

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Date: March 19, 2018

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 – ISSUANCE OF AMENDMENTS REGARDING THE ELIMINATION OF CERTAIN TECHNICAL SPECIFICATIONS REPORTING REQUIREMENTS (CAC NOS. MF9601 AND MF9602; EPID L-2017-LLA-0213) DATED MARCH 19, 2018

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